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

DEC 30 1992

Report No.: 50-302/92-28

Licensee: Florida Power Corporation
3201 -34th Street, South
St. Petersburg, FL 33733

Docket No.: 50-302 License No.: DPR-72

Facility Name: Crystal River 3

Inspection Conducted: November 30 - December 4, 1992

Inspector: for B. R. Crowley 12/22/92 Date Signed

Rich C. Chou
R. C. Chou 12/22/92 Date Signed

Approved by: J. J. Blake, Chief 12/22/92 Date Signed
Materials and Processes Section
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope:

This routine, announced inspection was conducted on-site in the areas of: Inservice Inspection (ISI), including repair and replacement (R&R) activities; implementation of Generic Letter (GL) 89-08 (Erosion/ Corrosion); in-process safety related pipe welding and nondestructive examination (NDE) activities; pipe support calculations and walkdowns; and licensee actions on previous inspection findings.

Results:

In the areas inspected, no deviations were identified. One non-cited violation, Failure to Follow Procedure for Calibration of MT Equipment - (paragraph 2.b,) and two new unresolved items, "Design Problem on Unistrut and Angle Supports," (paragraph 6.b,) and "Field Discrepancies on Pipe Supports," (paragraph 7,) were identified.

This inspection indicated that, in general, R&R activities were being accomplished in accordance with requirements. A good pipe welding program was in place and functioning well. With exception of the one violation noted above, the NDE program appeared to be good. Qualified personnel were

performing examinations in accordance with procedures in a conscientious manner. A detailed and comprehensive Erosion/Corrosion (E/C) program that continues to improve was in place. Problems still exist in the area of engineering analysis of as found conditions in pipe supports. Both the licensee and the AE have found problems with past support analyses and as-built installations.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*G. Cowles, Senior Nuclear Results Engineer
D. Czufin, Supervisor, Site Nuclear Engineering Services
F. Froats, Manager, Nuclear Compliance
*R. Frohnerath, Senior Mechanical Engineer - NOE
*H. Gelston, Acting Manager, Site Engineering Services
D. Gulling, ISI Specialist
*B. Gutherman, Nuclear Engineering Supervisor - NOE
*G. Halnon, Manager, Nuclear Plant Technical Support
*D. Jopling, Senior Nuclear Structural Engineer
*R. Knoll, Nuclear Engineering Supervisor - NCM
*D. Kurtz, Manager, Nuclear Operations Quality Assurance
*J. Lese, Senior Nuclear Structural Engineer
*C. Long, Supervisor, Nuclear Quality Control
*J. Maseda, Manager, Nuclear Operations Engineering
*B. McLaughin, Nuclear Regulatory Specialist
*A. Petrowsky, Supervisor, Site Engineering Services
J. Warren, Site Welding Engineer
K. Wilson, Manager, Nuclear Licensing

Other licensee employees contacted during this inspection included craftsmen, engineers, QC personnel, security force members, technicians, and administrative personnel.

NRC Resident inspectors

*P. Holmes-Ray, Senior Resident Inspector

*Attended exit interview

Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. Inservice Inspection (ISI) (73051) (73753)

See NRC Reports 50-302/92-05 and 50-302/92-14 for documentation of previous inspections in this area.

a. ISI Program Review (73051)

The inspectors reviewed documents and records as indicated below to determine whether ISI, specifically R&R activities, was being conducted in accordance with applicable procedures, regulatory requirements, and

licensee commitments. The applicable code for ISI is the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code, Section XI, 1983 Edition with Addenda through Summer 1983 with the extent of examinations of class 1 and 2 pipe welds determined by the requirements of the 1974 Edition with Addenda through the Summer 1975.

During the NRC inspection documented by Report 50-302/92-14, weaknesses were identified relative to the lack of licensee field overview and technical involvement in ISI activities. The licensee's ISI department was a part of the Reliability Centered Maintenance (RCM) organization. In response to this weakness, the licensee is reviewing their ISI program and organization to identify areas that need improvement and take appropriate corrective actions. As an initial step, the ISI organization now reports to the Manager of Nuclear Plant Technical Support.

b. Observation of Work and Work Activities (73753)

As noted in paragraph 3. below, leaks have developed in the Nuclear Services Closed Cycle Cooling (SW) system piping. The licensee was in the process of fabricating a new section of piping for replacement of the leaking section of pipe during the next outage (March, 1993). The inspectors examined the following R&R activities relative to fabrication of the replacement piping:

- Modification Approval Record (MAR) P-92-04-14-01 covers the replacement. The inspector reviewed the MAR package including:

  Work Request (WR) NU 020604 for shop fabrication

  WR NU 302697 for field installation

  ASME XI Repair/Replacement Program Work Request Evaluation

  ASME Section XI Code Reconciliation for WR NU 020604

These documents were detailed and provided good replacement suitability reviews in accordance with ASME Section XI. The Code reconciliation provided justification for using a later edition (1970 Addenda versus the 1969 Edition) of USA Standard (USAS) B31.7. The original construction code was USAS B31.1-67 with weld examination per USAS B31.7-69.

- The inspectors observed welding activities as follows:

  Completed weld surfaces were examined for welds SW-90-176 through 198.

  For welds SW-90-176, 177, 187, and 190, weld records consisting of Weld Traveler, Weld Inspection Plan, Welding Electrode Withdrawal Order, and Visual/Surface Inspection Report were reviewed.

In-process welding activities, including in-process welding and inspection records, were observed for weld SW-90-199. Welder qualification records for welders NW-66, NW-147, NW-171, NW-218, NW-309,and NW-372 were reviewed.

Welding material certification records for 3/32" ER70S-2, heat number L02015, and 3/32" E7018, heat number 37628, were reviewed.

Weld appearance, weld material control, use of correct weld material, use of correct welding procedure, welder qualification and general knowledge and ability, weld material certification, welding procedure qualification, and use of code required weld inspection methods and extent were examined.

Based on inspection of the above welding activities, it appeared that a good welding program was in place and was being properly implemented in a conscientious manner by qualified personnel.

- The inspectors observed Magnetic Particle (MT) inspection for welds SW-90-196, 197, 198, and 199. The observations were compared with the applicable procedures and the applicable code in the following areas: examination methods; contrast of dry powder particle color with background; surface temperature; suspension medium for wet particles, if applicable; viewing conditions; examination overlap and directions; pole or prod spacing; current or lifting power (yoke); and acceptance criteria.

In addition, the MT procedures, Special Process Specification (SPS) MT-N01, Revision 1, Dry Visible Magnetic Particle Examination Technique and SPS MA-N01, Revision 1, Magnetic Particle Acceptance Criteria USAS B31.7 were reviewed. NDE certification and qualification records for two NDE MT examiners were examined.

During these observations and reviews, the inspectors identified a problem with calibration of the MT Yoke relative to the proper pole spacing. SPS MT-N01 specifies that the maximum pole spacing be determined by the ability of the yoke's lifting power and the pole spacing be documented on the calibration record. When questioned by the inspector, the licensee NDE examiner stated that the maximum pole spacing allowed was 8" since that was used during calibration. A review of the calibration record revealed that the pole spacing was not recorded on the calibration record. Further investigation revealed that the Calibration Lab calibrates MT equipment in accordance with a Calibration Lab procedure and records the calibration on a standard Calibration Lab data sheet rather than in accordance with SPS CT-N05 as required by SPS MT-N01. Therefore, the maximum pole spacing was not recorded as required by SPS MT-N01 and SPS CT-N05. Discussions with NDE and Calibration Lab personnel revealed that a pole spacing of 8" has always been used for calibration and it is standard industry practice and licensee practice to use an 8" maximum pole spacing. Therefore, it appears this procedural violation did not affect actual testing.

Problem Report (PR) 92-9094 was issued immediately and corrective action to change procedures to implement a more workable calibration and calibration documentation process, was initiated. In addition, QC physically verified that the yoke used for the above examinations would successfully lift the required weight at a pole spacing of 8". Although failure to follow procedures for documenting MT yoke pole spacing is in violation of licensee procedures and 10 CFR 50, Appendix B, Criterion V, the violation did not affect the actual calibration or MT test. This appeared to be an isolated case and immediate corrective action was initiated. This NRC identified violation is not being cited because criteria specified in section VII.B of the NRC Enforcement Policy were satisfied. This violation is identified as NCV 302/92-28-01, Failure to Follow Procedures For Documenting Pole Spacing During Calibration of MT Yokes.

With exception of the one violation, the inspection indicated the NDE program was working well; inspections were being performed in a conscientious manner by qualified personnel. R&R activities were being controlled in accordance with ASME Section XI and licensee procedures. One non-cited violation, as detailed in 2.b above, was identified. No deviations were identified.

3. Nuclear Services Closed Cycle Cooling System (SW) Piping Leaks

On November 6, 1992, the licensee requested approval, under Generic Letter 90-05, to perform a temporary non-code repair to a leak in the SW piping system. The location of the leak was in the 6" section of the SW pump suction piping downstream of the surge tank adjacent to the 6" to 18" transition piece. The leak was approximately 30 drops per minute. A code repair was considered impractical since the repair would require the removal of the entire SW system from service for an extended period of time to drain and repair or replace the affected piping. The cause of the leak was exterior corrosion caused by periodic standing sea water in the trench the piping is located in. The flaw could not be fully characterized because of the general nature of the corrosion and potential to increase the leakage if surface preparation were attempted for characterization by NDE. The method of temporary repair was to encase the leaking area in concrete. The temporary repair was approved by the NRC and implemented. Shop fabrication of a new 18" header and 6" piping was being completed during this inspection. Installation of the new piping is planned for the March 1993 outage.

During further licensee inspections, two additional leaks, similar to the first, were identified in the 18" header. The leaks were at the north end of the suction header between the two suction risers for SW pump "A". On November 13, 1992, the licensee requested approval for a temporary non-code repair of these leaks using pipe clamps and neoprene seals over each leak. The licensee assessed the overall condition of the piping by visual and Ultrasonic UT thickness measurements. These inspections indicated a fairly uniform corrosion of the full circumference of the 6" pipe and the lower one third of the circumference of the 18" pipe. Specification nominal wall thickness for

the piping is 0.375". The general remaining thickness of the 6" pipe was found to be 0.070" to 0.080". For the bottom one third of the circumference of the 18" piping, the general remaining thickness was found to be 0.060" to 0.080". The licensee evaluated the piping to the design code, USAS B31.1, using reduced wall thicknesses as defined by the UT thickness measurements. Based on calculations, the licensee determined that the piping meets ANSI B31.1 with the significantly reduced wall thicknesses. This condition and the November 13th request for approval of a temporary non-code repair were still being evaluated by NRR at the time of this inspection.

On November 30, 1992, the licensee provided additional information to the NRC relative to plans for monitoring the piping until replacement during the March 1993 outage.

The following summarizes the inspector's observations and reviews during the current inspection:

- The inspectors observed the piping in the plant and reviewed the UT thickness measurement. In addition, a video of the underside of the 18" pipe, including the leaks, was reviewed. The observations verified that the full circumference of the 6" pipe and the bottom one third of the circumference of the 18" pipe have general corrosion deterioration of the outside surfaces.

- PR 92-0168, including the preliminary action plan was reviewed. The cause of the pipe corrosion was attributed to allowing sea water, during maintenance of the SW and DC heat exchangers, to drain directly on the 6" pipe and remain in the trench containing the 18" and 6" pipe. In addition, the auxiliary building sump float level control was not set properly to keep water from standing in the trench and coming in contact with the pipe. These conditions have been corrected. As outlined in the November 30, 1992, letter to the NRC and the PR action plan, the following monitoring is in place:

  Routine UT thickness measurements will be performed on a 45 day frequency.

  The trench will be vacuumed as necessary to keep the underside of the pipe dry.

  Operators will monitor the SW header once each shift to detect signs of increasing leakage.

  Salt water from maintenance will be drained directly to the sump and operations will be cautioned not to exceed the capacity of the sump when draining the heat exchangers.

As noted above, NRC is still reviewing this issue. No violations or deviations were identified.

4. Feedwater (FW) System Leak

Based on sump volume increases, the licensee has investigated and determined that the FW system is leaking inside the reactor building. The most likely location has been determined to be one of the flanged nozzle connections at the B steam generator. This problem is identified on PR 92-0121. The inspector reviewed this problem with the licensee. The following summarizes the review:

- The current leakage rate is approximately 4 gallons per minute (GPM) and is increasing at a rate of 0.76 GPM per month. If the leak continues to increase at this rate, the licensee has concluded that the leak will be at the limit set by them of 6 GPM about March 1, 1993. The 6 GPM limit is based on the licensee's capacity for processing the water. As noted above, the outage is now planned for March, 1993.

- The inspectors reviewed PR 92-0121, including the action plan and meeting minutes for planning the repair. In addition, daily plots of leakage data were reviewed.

The above reviews revealed that the licensee is closely monitoring the leak to determine if early shutdown to repair the leak is warranted. The PR and action plan, including repair plans were found to be detailed and comprehensive. No violations or deviations were identified.

5. Erosion Corrosion (E/C) (Generic Letter 89-08)

See NRC Report 50-302/92-14 for documentation of a previous inspection in this area.

During the inspection documented by Report No. 92-14, the inspectors identified the following two weaknesses with licensee Preventive Maintenance (PM) Procedure No. PM-251 "Turbine Piping Systems Erosion/Corrosion Inspection".

Enclosure 4, which describes the grid layout guidelines did not implement the recommendations of M87-1006, EPRI, or the draft ASME Subsection IWH for extending the examination of a component upstream and downstream of the component's welds.

Enclosure 4, paragraph 8.g, stated that,"Not all points defined by the grid will be inspected, but the whole elbow must be laid out to ensure repeatability". This is not in accordance with EPRI's recommendations or the draft ASME Subsection.

These weaknesses were corrected in revision 5 of procedure PM-251. In addition to review of the procedure changes, the inspectors discussed the E/C program with licensee personnel. The following describes various aspects of the licensee's program:

- The licensee's method of predicting which systems are most susceptible to E/C include EPRI Report NP-3944, Keller's Method, and

EPRI CHECMATE Codes. The current program is based on EPRI Report NP-3944 and Keller's Method. However, the licensee has recently obtained CHECMATE computer codes and has started to model their systems using these codes and comparing the results with previous results from their current program. FW train "A" inside the reactor building has been modeled with CHECMATE. For this system, the most susceptible components from the CHECMATE modeling agree with the most susceptible components from the current program (Keller). The ranking of other less susceptible components does not agree. The licensee plans to model all large bore susceptible systems with CHECMATE. Preliminary plans are for this to be accomplished in the second half of 1993 and the first half of 1994. Also, future plans are to use CHECKWORKS (CHECK-NDE, CHEC-T, and CHECMATE) and compare the results with the results from the current program.

The Licensee's E/C program includes the systems that have been predicted to be susceptible to E/C. The model ranks each system's components according to their E/C susceptibility.

- The licensee's primary basis for selecting components for UT inspection at the next outage is the results from the predictive analysis. Components are re-inspected at 50% of their predicted life. Engineering judgement, industry experience and results from previous UT inspections are used to select supplemental inspection locations.

- The following systems are included in the licensee's E/C program:

  Feedwater
  Condensate
  Feedwater Heater Drains
  Moisture Separator Drains
  Moisture Separator Reheater Drains
  Extraction Steam
  Steam Generator Blowdown
  Auxiliary Steam (wet areas only)
  Crossover and Crossunder Piping (Crawl through inspection)
  Main Steam traps are routinely replaced as maintenance items.

- The following components are included in the E/C program:

  Control Valves
  Tees and Branches
  Expanders and Branches
  Flow Nozzles and Orifices
  Elbows and Reducing Elbows
  Straight Pipe Sections
  Some Heater Drain Tanks

- In addition, exit nozzles for some tanks have been inspected, although not in the program.

- The following criteria is used by the licensee for excluding systems from the program:

  Systems with operating temperature and pressure under 200 degrees F and less than 275 psi, respectively, are excluded.

  Stainless steel systems are excluded.

  Systems containing superheated steam are excluded.

- UT thickness measurement procedures are approved in accordance with the licensee QA program. Procedures define methods for: performing UT measurements, grid spacing criteria, ensuring that followup inspections are taken at the same locations as the previous inspections, ensuring the use of certified inspection personnel, analyzing the results of UT measurements, and establishing acceptance criteria.

- Minimum wall thickness requirements for safety-related and non-safety related piping are based on code allowable minimum wall thickness.

- The licensee's method of analyzing UT data includes: calculating the wear rate (wr) based on point to point measurements, band method and judgement; predicting the thickness (t) at the next refueling outage based on $t_{predicted} = t_{measured}$ - wr X current operating cycle time; determination of current acceptance based on $t_{measured} > t_{minimum}$ and $t_{predicted} > t_{minimum}$; and calculation of remaining life based on remaining life = $t_{measured} - t_{minimum}$ divided by the wr.

- Repair and replacement of safety-related eroded components is covered by the licensee's R&R program meeting ASME Section XI. Repair and replacement of Balance of Plant components is covered by the licensee's WR system and welding and NDE of repairs and replacements are accomplished by qualified personnel and suitable procedures (same welding and NDE program as used for safety-related piping).

- Site Engineering has responsibility for the E/C program. The program is covered by the licensee's QA program.

Based on the above review, the inspectors concluded that the licensee has a detailed and comprehensive E/C program that is continually updated to meet industry practices and standards. Within the area examined, no violations or deviations were identified.

6. Review of pipe Support Calculations

a. Licensee Event Report (LER) 92-06, Evaluation Oversight Causes Wej-It Pipe Supports Factor of Safety to be Less Than NRC Requirements.

On May 1, 1992, with Crystal River Unit 3 (CR-3) in Mode 5 (Cold Shutdown), the licensee reported to NRC that four safety-related pipe supports did not meet the minimum factors of safety specified in NRC IE Bulletin 79-02, Revision 2. The licensee declared support DHH-646 in DHP-14 suction line, inoperable, resulting in the "A" train Decay Heat (DH) being declared inoperable. Within 72 hours, the DH support was repaired and the system was returned to service. The remaining three supports and their systems were not required in Modes 5 and 6, but they were repaired before the end of the last refueling outage.

The licensee's Architect-Engineer (AE), Gilbert/Commonwealth, Inc., (G/CI) discovered the problem in late 1991 during analysis of a section of piping for a future modification. During this analysis, G/CI discovered that the reduction in allowable loads for Wej-It anchor bolts did not consider the previous allowable load reduction due to the base plate prying action, for anchor bolts with factors of safety between 4 and 8 in the original calculations.

Anchor bolt prying exists where the base plate corners tend to add to the tension load in the anchor bolts by applying a prying force to the bolts. In 1979, during the original review of safety factors for IE Bulletin 79-02, G/CI used a 50% anchor bolt capacity reduction to allow for prying action. At that time, G/CI reviewed, revised, and documented the calculations with factors of safety below 8.0 for anchor bolt prying action. The calculations with a factors of safety above 8.0 (included) were not reviewed since the final factor of safety would be 4.0 or above, and would meet the IE Bulletin 79-02 requirements.

In 1984, G/CI reviewed the pipe support calculations of Wej-It anchor bolt and applied a 50% capacity reduction based on manufacturer's notice and the site test results. During this review, G/CI only reviewed the anchor bolt calculations with a factor of safety below 8.0 and above 4.0. Original calculations with a factor of safety above 8 were not reviewed since G/CI engineers considered that an original factor of safety above 8.0 would be reduced to above 4.0 with the Wej-It anchor bolt 50% capacity reduction. The G/CI engineers had apparently forgotten about the 50% safety factor reduction to be applied because of prying action, which when applied would have reduced original factors of safety in the range of 8.0 to 16.0, to 4.0 to 8.0, which would have required corrective action when the additional 50% safety factor reduction was applied to the Wej-It anchor bolts capacity.

G/CI initiated a sampling program to review 55, randomly selected, pipe supports and found a support with a final factor of safety of 2.0 (page 2 - Attachment to FCS-12772, January 6, 1992). Based on the sampling procedure, an additional 30 supports were randomly selected and reviewed (page 5 - Attachment to FCS-13052, May 4, 1992). The results of the evaluation of were that 11 of the 30 supports did not meet the factor of safety of 4.0 required by IE Bulletin 79-02. Four of the eleven pipe supports, DHH-646 , MUH-517, SWH-104, and SWH-267 possessed anchor bolt safety factors that did not meet the short term operability requirements of IE Bulletin 79-02, (i.e., factor of safety less than 2.) The

remaining seven pipe supports met the short term operability of IE bulletin 79-02, but required upgrade for long term plant operability. Based on the above analysis, G/CI informed the licensee about the results and the licensee declared support DHH-646 and its system inoperable on May 1, 1992, which resulted in LER 92-06.

Due to the failure of the 11 supports, G/CI reviewed the remaining sample of 249 supports by June 21, 1992. Forty-three of the 331 supports reviewed were determined to have factors of safety below 4.0, with 35 between 2.0 and 4.0, five (MUH-517, MUH-628, SWR-519, SWH-024, and SWH-267) between 1.0 and 2.0, and two supports (DHH-646 and SWH-104) with factors below 1.0. The two with factors of safety below 1 were repaired before the end of the last refueling outage (April - July, 1992).

By December 1, 1992, by removing some analytical conservatisms and using computer analysis, the licensee was able to reduce the number of supports to be repaired to 17 supports with factors of safety between 2 and 4, six supports with factors of safety between 1 and 2 (5 of the 6 have been repaired), and three supports with factors of safety below 1 (2 of the 3 have been repaired). All of the supports requiring modifications are scheduled to be repaired by the end of the next outage which starts in March, 1993.

The LER remains open pending further review of licensee actions during the upcoming outage.

b. PR 92-0177, Unistrut Support Design Problems

This problem report was written on November 12, 1992, based on G/CI letters FCS-13419, dated November 3, 1992 and FCS-13454, dated November 12, 1992. A re-analysis of piping stress analysis CR-77, was performed by G/CI to correct a minor error in the piping configuration and to change a local coordinate system to a global coordinate system. Although pipe stresses were found to be acceptable and new support loads were only slightly higher than the previous design loads, 17 of 43 supports were found to have safety factors of less than the design values of 4.0 for Operating Basis Earthquake (OBE) or 2.0 for Safe Shutdown Earthquake (SSE) when consideration was given to eccentric loads which were not considered in the original design calculations.

Five supports, designed with embedded unistruts, (DHH-501, 502,503, 507,and 518 had factors of safety less than 1.0 for combined dead weight and seismic loadings. Three of these supports, (DHH-501, 503 and 507) had factors of safety less than 1.0 for dead weight loading alone.

The licensee considered that the plant was capable for continued operations based on four factors listed in FCS-13454, as shown below:

- Conservatisms in the prying factor were used to analyze these supports by hand calculations. The prying force would be reduced if a finite element analysis such as PRYTEN were used.

- Preliminary walkdowns of the piping and supports revealed no evidence of support failures or deterioration.

- The Unistrut Corporation provided ultimate loads with a lower bound of 6700 lbs. and an upper bound of 8800 lbs. based on test results; while an ultimate load of 6400 lbs. was used in the design calculations for Unistrut's P-3200 series.

- The reduced operating temperature of 120 degrees F could be used for the stress analysis instead of 160 degrees F used for the previous analysis. Two percent damping value can be used for the evaluation of the SSE condition versus the one percent previously used.

After removing the conservatisms such as high temperature, high damping value, high seismic factor, etc., and using a finite element analysis, the licensee calculated the final factors of safety for five supports, previously considered to have factors of safety less than 1.0, to be above 1.0 with a lowest of 1.09 for support DHH-502. Therefore, the licensee considered all of the supports to have a factor of safety above 1.0 and all of the systems are operable.

The inspectors discussed the problem with the licensee's engineers and reviewed the information provided. The inspectors partially reviewed calculation Nos. DC-5520-100.1PE, DC-5520-132.1PE, and DC-5520-132.3PE.

Calculation No. DC-5520-100.1PE was a preliminary calculation which identified 15 of 43 supports to have factors of safety below 4.0 for OBE or below 2.0 for SSE. Five of the 15 supports, (DHH-501, DHH-502, DHH-503, DHH-507, and DHH-518) were listed as having factors of safety below 2.0 for OBE or below 1.0 for SSE, which did not meet operability requirements. This was the primary calculation for the plant operating concern and was the basis to generate PR No. 92-0177 to describe the problem on the design of Unistrut supports.

Calculation DC-5520-132.1PE was to provide the necessary backup technical justification for the re-design of the five pipe supports which did not meet operability requirements in the preliminary calculation. These supports will meet the long term operating requirement after redesign and installation. The engineering packages for these supports will be issued to the site for modification as soon as possible.

Calculation No. DC-5520-132.3PE is a calculation, refined by removing the conservatisms, and analyzed with the finite element method, to prove the final factors of safety for the five supports meeting the operability requirements . The lowest factor of safety for the five supports from this analysis was 1.09 for SSE for support DHH-502.

Therefore, all the supports in the DH system met the plant operability requirements.

Pending the licensee's modification of this DH system, root cause investigation and prevention, and review for other systems containing angle and unistrut supports, this item is identified as Unresolved Item 302/92-28-02, Design Problem on Unistrut and Angle Supports.

c. Review of Hilti Anchor Bolt Allowable Loads

The Hilti Expansion Anchor Bolts, Kwik Bolt II and HSL Metric Heavy Duty, were used at CR. The design allowable loads identified in "Crystal River 3, Hilti Expansion Anchors, SP-88-005, Final Design Guide and Installation Instructions," dated September 11, 1989, were higher than the allowable loads stated in the Hilti design catalog, "Hilti Fastening Technical Guide", Published in 1991 by Hilti, Inc. As an example, the allowable loads for HSL Metric Heavy Duty Expansion Anchor Bolts given in the licensee's design guide are from 13 percent to 61 percent, (with a majority around 50 percent,) higher than the Hilti design catalog.

The basis for establishing higher allowable loads than the loads in the Hilti design catalog was that test results conducted at the Crystal River site produced higher results than the average of the results from tests conducted by Hilti, Inc. at other locations or plant sites. The tests performed at Crystal River were also conducted by Hilti, Inc., who issued a test report to FPC which was used as the basis for generating the higher design allowable loads. The differerce in test results between the CR site and other locations could be the difference between the strength of newly poured concrete and the existing concrete. Concrete strength is tested and specified based on a curing time of 28 days. The concrete tends to increase in strength with time.

Near the end of the inspection, the licensee provided calculation No. DC-552154.1-SE, which is a basis for establishing the design allowable loads, based on the site test results, for review. There were no immediate questions with the information provided, but this subject will be reviewed further during future inspections.

Within the areas inspected, no violations or deviations were identified.

7. Pipe Support Walkdown Inspections

Per PR 92-0177, noted above, which describes the current pipe support conditions, the licensee's engineers who walked down the line and inspected the supports did not find any cracked welds, bent or deformed members, or other irregularities on the pipe supports.

To verify the current pipe support conditions, and compare the as-built installation with the as-built drawings, the inspectors, with assistance

from licensee engineers, walked down the supports to inspect for damages and take measurements in the field. The supports were partially inspected against as-built drawings for such things as configuration, identification, fasteners/anchor installation, anchor size, marking, base plate size and thickness, plate warpage, member sizes, dimensions, interferences, cracks in welds, bent or deformed members, bolt edge distances, bolt hole sizes, oxidation accumulation, and general maintenance.

The supports inspected during the current inspection are listed below:

Walkdown Inspection Of Supports

Item No.	Support No.	Rev. No.	Discrepancies/Comments/ Licensee Remedies
1.	DHH-501	2	The two 4"X4"X1/2" angles attached to the embedded unistrut had oversized or slotted holes on the east end connections.
2.	DHH-502	2	It appeared that the C6X8.2 horizontal channel section might have a slight gap at the concrete wall interface. (The inspector examined this support at a distance because the ladder could not be set at the support location.)
3.	DHH-503	2	The lower 4"X4"X1/2" angle attached to embedded unistrut had an oversized or slotted hole.
4.	DHH-504	3	None
5.	DHH-507	2	A one inch gap was found between the nut and clamp ear on the clamp load bolt. It appeared that a wrong bolt was used without enough thread to tighten the nut. The washer plate, used to carry the rod load, Item E in the Material Bill List, was found to have 1/4" net edge distance (between edges of the hole and member) for 1" dia. bolt.
6.	DHH-508	2	The washer plate, used to carry the rod load, Item E in the Material Bill List, was found to have 1/4" net edge distance (between edges of the hole and member) for 1" dia. bolt.
7.	DHH-510	N/A	The main rod of the support was in contact with an adjacent diagonal unistrut brace for a conduit support.

Walkdown Inspection Of Supports (continued)

Item No.	Support No.	Rev. No.	Discrepancies/Comments/ Licensee Remedies
8.	DHH-511	5	A 3/16" fillet weld was found all around the connection between a wide flange and the base plate. The drawing required a 1/4" fillet on both sides of the web. 3/16" and 1/8" fillet welds were found at connections between west and east stop angles and the wide flange beam. The drawing requires both to be 1/4" fillet welds.
9.	DHH-514	4	The same discrepancy as the first item for DHH-511 above. In addition, a 3/16" fillet weld was found at connections between both the west and east stop angles and wide flange beam; the drawing requires both to be 1/4" fillet welds.
10.	DHR-6	3	Eight, 3/4" dia. Phillips wedge anchors were found to be installed instead of the 3/4" dia. Wej-It type anchors shown on the drawing. The licensee will revise the drawing and calculations to show the correct anchor type.
11.	DHR-14	2	None

During the inspection of the line and supports, the inspectors did not find any irregularities such as cracked welds or bent or deformed members. However, as noted above, discrepancies were noted when comparing the as-built supports to the as-built drawings.

The licensee's resolution of these discrepancies appeared to be questionable to the inspectors. The resolution referred to the Manual of American Institute for Steel Construction (AISC) and considered all of the discrepancies as insignificant to safety. As examples: The engineers interpreted AISC minimum bolt edge distance requirements to be applicable only to high strength bolts and not to the CR installed support bolts. (The inspectors consider this to be an incorrect interpretation of AISC requirements.) The engineers did not consider the gap between the nut and the clamp ear on support no. DHH-507 to be significant. In addition, the engineers did not consider the need to remove the interference between the pipe support, DHH-510, rod and the conduit support, or to investigate and revise documents for the undersized welds in six locations on supports DHH-511 and DHH-514. Pending the licensee's re-review and investigation of the noted

discrepancies, this matter will be identified as an unresolved item, 302/92-28-03, Field Discrepancies on Pipe Supports.

In the areas inspected, no violations or deviations were identified.

8. Licensee Actions on Previous Inspection Findings

a. (Open) VIO 302/92-14-01, Inadequate Procedure and Drawings for ISI Activities.

This violation involved: (1) generic use of Babcock and Wilcox Nuclear Services (BWNS) UT procedure, ISI-119, Rev.11, without controlling the equipment and test parameters so that examination sensitivity could be verified and repeated, and (2) the failure of ISI calibration block drawings to reflect the as-built condition of the blocks. The licensee's letter of response is dated August 21, 1992. The letter of response has been reviewed and found to be acceptable. The inspectors reviewed the status of corrective actions as detailed below.

The licensee attributed the violations to weaknesses in the ISI program that, for item (1), did not require a technical review of vendor ISI procedures by the licensee prior to implementation by the contractor and, for item (2), did not provide specific guidelines for receipt inspection or proper handling of calibration blocks.

For item (1), the licensee took immediate corrective action to revise the procedure in question at the time the violation was identified. In addition, the procedure has been reviewed by EPRI. Their comments are presently being reviewed and the procedure will be revised as appropriate. Relative to item (2), at the time of this inspection, as-built drawings had been obtained or generated for all calibration blocks. In addition to these specific corrective actions, a number of improvements have been made or, are planned for the ISI program. The licensee plans to evaluate their ISI organization and make changes as necessary. As noted in paragraph 2. above, one improvement, i.e., moving the ISI organization from the Maintenance Department to Site Engineering Support, has been accomplished. Other corrective actions planned include revision of the program to require technical review of ISI procedures by the licensee and receiving inspection of ISI calibration blocks. Full corrective actions are not scheduled for completion until the next refueling outage or, April of 1994.

Corrective actions will be reviewed further during future inspections.

b. (Closed) VIO 302/92-14-02, Failure to initiate appropriate Corrective Action for Identified Equipment Deficiencies.

This violation involved failure to initiate appropriate corrective action for ASME Class 1 valve bolting deficiencies identified during ISI inspections. The licensee's letter of response was issued on August 21,

1992. The inspectors reviewed the letter of response and found it to be acceptable.

The following summarizes the licensee's corrective actions and reviews by the inspectors:

The licensee attributed the violation to personnel error. The corrective actions included issuing a PR for the problem in question and counseling the personnel relative to the need to issue PRs in a timely manner. In addition, independent of this violation, the licensee is in the process of improving the clarity and consistency of the PR generation and review process. The violation response indicated an action plan for this effort would be in place by September 30, 1992.

The inspectors verified, by interviews, that personnel had been counseled. In addition, the inspectors reviewed a letter to all Nuclear Plant Systems Engineering (organization personnel responsible for the violation report to) emphasizing the requirements for issuing PRs and issuing them in a timely manner. The inspectors also reviewed the action plan noted above and noted that the plan includes a recommendation for training on the PR process for all personnel. In addition the action assignment matrix for the Cooperative Management Auditing Program (CMAP) audit assigns an action and responsibility for training all departments in the PR process.

The inspectors also reviewed PR 92-0048 for the valve bolting problems identified in the violation. The PR was detailed and corrective actions resulted in the licensee identifying a number of discrepancies in bolt torque values used for valve maintenance. Corrective actions appear to be adequate.

Based on the above reviews of the licensee's corrective actions, this item is closed.

c. (Closed) UNR 302/92-14-03, Loose Pipe Hanger Components.

This item identified two pipe supports (SWH-340 and MUH-79) with loose components. Neither hanger supported any dead weight load. PR 92-051 was issued to investigate this condition and document any required corrective actions. The licensee's inspection found that the supports in question were out of adjustment. Corrective actions included: (1) tightening and adjusting the supports in question, (2) an assessment of the redistributed loads while the supports were out of adjustment, and (3) assessment of the ability of the supports to function as seismic restraints in their loose conditions. WR NU 0298235 was issued to tighten and adjust the two supports. Ability of the supports to provide adequate seismic support and the affects of the redistribution of dead loads to the adjacent supports were evaluated with calculation S-92-0128. The calculation showed that the supports would have provided adequate seismic restraint had it been necessary and the piping and

adjacent supports were not adversely affected by the lack of dead weight support by the two support.

The inspectors reviewed the above corrective actions. Other supports were inspected during this inspection and at the time of the original inspection findings. No other supports with loose components were identified. This item is closed.

d. (Open) IFI 302/91-01-03, Adequacy of Small Bore Pipe Hangers.

This item concerned deficiencies such as loose clamps, wrong components used, etc., identified on small bore pipe supports. Questions were raised relative to the as-built condition differing from the as-built drawings. The licensee stated that the small bore piping and supports had never been "as-built" because the supports were installed per "typical" sketches and located on piping isometric drawings. The inspectors discussed the concern with licensee engineers and reviewed the information provided.

The licensee provided Gilbert/Commonwealth, Inc. letter FCS-6940, dated October 15, 1985, for review. The letter contained a summary report for CR Unit 3 for Wej-It Expansion Anchor Evaluation on Small Bore Pipe Supports. The report indicated that five small bore piping systems with 67 supports were selected for as-built walkdowns and rigorous computer analysis to evaluate the adequacy of the Wej-It expansion anchors in small bore piping. This inspection and analysis was performed because site test results found that the design capacity of Wej-It expansion anchors was 50% of the rated capacity. (Note: Each support in safety-related large bore piping systems was reviewed individually per IE Bulletins 79-02 and 79-14 requirements).

Of the 67 supports reviewed by the licensee, only one was found to have a factor of safety (FS) less than 4.0. The FS for this support was only 2.85 due to the as-found support having a missing anchor bolt. With the bolt installed in accordance with the design intent, the FS is 12.2. The total results were: nine supports had factors of safety greater than 4 but less than 8; 18 supports had factors of safety greater than 8 but less than 24; and 40 supports had factors of safety greater than 24. This report showed that the small bore piping systems installed in the field were acceptable, based on the typical design drawings, even with some deficiencies existing.

Since this report was completed in 1985, before LER 92-06 and PR 92-0177 (discussed in paragraph 6 above), the support calculations might not have considered bolt prying action or eccentricity load problems on the unistrut supports. Pending licensee review of the Gilbert/Commonwealth report based on recent discoveries of prying action and eccentricity load problems, this item will remain open.

In the areas inspected, no violations or deviations were identified.

9. Exit Interview

The inspection scope and results were summarized on December 4, 1992, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results listed below. No proprietary information is contained in this report. Issues relative to the two UNRs listed below were identified to the licensee as concerns. On December 7, 1992, the inspectors informed the licensee, K. Wilson, by telephone, that the issues would be identified as URIs pending further NRC inspections.

(Closed) NCV 302/92-28-01, Failure to Follow Procedure for Calibration of MT Equipment - Paragraph 2.b

(Open) UNR 302/92-14-02, Design Problem on Unistrut and Angle Support - Paragraph 6.b

(Open) UNR 302/92-14-03, Field Discrepancies on Pipe Supports - Paragraph 7

10. Acronyms and Initialisms

AE	-	Architect Engineering
AICS	-	American Institute For Steel Construction
ASME	-	American Society of Mechanical Engineers
B&PV	-	Boiler and Pressure Vessel
BWNS	-	Babcock & Wilcox Nuclear Services
CR	-	Crystal River
CMAP	-	Cooperative Management Auditing Program
DH	-	Decay Heat System
E/C	-	Erosion/Corrosion
FPC	-	Florida Power Corporation
FW	-	Feedwater System
G/CI	-	Gilbert/Commonwealth Inc.
GL	-	NRC Generic Letter
GPM	-	Gallons Per Minute
ISI	-	Inservice Inspection
LER	-	Licensee Event Report
MAR	-	Modification Approval Record
MT	-	Magnetic Particle
NCV	-	Non-cited Violation
NDE	-	Nondestructive Examination
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
OBE	-	Operating Basis Earthquake
PM	-	Preventive Maintenance
PRC	-	Plant Review Committee
psi	-	Pounds Per Square Inch
QA	-	Quality Assurance
QC	-	Quality Control
R&R	-	Repair and Replacement
RCM	-	Reliability Centered Maintenance

SNES	-	Site Nuclear Engineering Services
SPS	-	Special Process Specification
SSE	-	Safe Shutdown Earthquake
SW	-	Nuclear Services Closed Cycle Cooling System
UNR	-	Unresolved Item
t	-	Thickness
USAS	-	USA Standard
UT	-	Ultrasonic Test
WR	-	Work Request
wr	-	Wear Rate