Official

December 4, 1990

Docket No. 50-261
License No. DPR-23

Carolina Power and Light Company
ATTN: Mr. Lynn W. Eury
Executive Vice President
Power Supply
P. O. Box 1551
Raleigh, NC 27602

Gentlemen:

SUBJECT: ERRATA LETTER FOR NRC INSPECTION REPORT NO. 50-261/90-24

The subject inspection report was transmitted to you by our letter, dated November 20, 1990. It has come to our attention that the Notice of Violation, was not transmitted with the report details. We have enclosed a copy of the Notice of Violation. Please insert the pages into your copy of the report.

If you have any questions concerning this matter, please contact us.

Sincerely,

Original signed by
Caudle A. Julian

Caudle A. Julian
Engineering Branch
Division of Reactor Safety

Enclosure:
Copy of Notice of Violation

cc w/encl:
C. R. Dietz, Manager
Robinson Nuclear Project Department
H. B. Robinson Steam Electric Plant
P. O. Box 790
Hartsville, SC 29550

J. J. Sheppard, Plant General Manager
H. B. Robinson Steam Electric Plant
P. O. Box 790
Hartsville, SC 29550

(cc w/encl cont'd - See page 2)

IE01

(cc w/encl cont'd)
Heyward G. Shealy, Chief
Bureau of Radiological Health
Dept. of Health and Environmental
Control
2600 Bull Street
Columbia, SC 29201

Dayne H. Brown, Director
Division of Radiation Protection
N. C. Department of Environment,
Health & Natural Resources
P. O. Box 27687
Raleigh, NC 27611-7687

McCuen Morrell, Chairman
Darlington County Board of Supervisors
County Courthouse
Darlington, SC 29535

Richard E. Jones, General Counsel
Carolina Power and Light Company
P. O. Box 1551
Raleigh, NC 27602

H. A. Cole
Special Deputy Attorney General
State of North Carolina
P. O. Box 629
Raleigh, NC 27602

Robert Gruber
Executive Director
Public Staff - NCUC
P. O. Box 29520
Raleigh, NC 27626-0520

J. D. Kloosterman, Director
Regulatory Compliance
H. B. Robinson Steam
Electric Plant
P. O. Box 790
Hartsville, SC 29550

(bcc w/encl - See page 3)

bcc w/encl:
Document Control Desk
H. Christensen, RII
R. Lo, NRR

NRC Resident Inspector
U.S. Nuclear Regulatory Commission
Route 5, Box 413
Hartsville, SC 29550

RII:DRS
JLC
JLColey:td
12/4/90

RII:DRS
JJBlake
12/4/90

ENCLOSURE 1

NOTICE OF VIOLATION

Carolina Power and Light Company
H. B. Robinson

Docket No. 50-261
License No. DPR-23

During an NRC inspection conducted on October 29 - November 2, 1990, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1990), the violation is listed below:

10 CFR 50, Appendix B, Criterion V as implemented by Carolina Power and Light Company Corporate Quality Assurance Manual, Section 6, paragraph 6.3.2 requires that, the accomplishment of activities affecting quality shall be in accordance with approved procedures and/or drawings which are appropriate to the circumstances.

Contrary to the above, on October 3, 1990, the inspector observed that an inservice inspection isometric sketch (No. CP&L-234B) did not accurately depict the actual pipe configuration in that, three field welds were not shown for the piping run illustrated. Subsequent review by the licensee established that similar errors existed on other isometric sketches. Inaccurate isometric sketches could result in an inadequate total weld population subject to inservice inspection and could result in nondestructive examiners performing inservice inspections on incorrect welds.

This is a Severity Level IV violation (Supplement I.D.3).

Pursuant to the provisions of 10 CFR 2.201, Carolina Power and Light Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region II, and if applicable, a copy to the NRC Resident Inspector, H. B. Robinson within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order may be issued to show cause why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

FOR THE NUCLEAR REGULATORY COMMISSION

original signed by
Jerome J. Blake

Caudle A. Julian, Chief
Engineering Branch
Division of Reactor Safety

Dated at Atlanta, Georgia
this 20th day of November 1990

Official

November 20, 1990

Docket No. 50-261
License No. DPR-23

Carolina Power and Light Company
ATTN: Mr. Lynn W. Eury
Executive Vice President
Power Supply
P. O. Box 1551
Raleigh, NC 27602

Gentlemen:

SUBJECT: NOTICE OF VIOLATION
(NRC INSPECTION REPORT NO. 50-261/90-24)

This refers to the inspection conducted by J. L. Coley of this office on October 29 - November 2, 1990. The inspection included a review of activities authorized for your H. B. Robinson facility. At the conclusion of the inspection, the findings were discussed with those members of your staff identified in the report.

Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress.

The inspection findings indicate that certain activities appeared to violate NRC requirements. The violation, references to pertinent requirements, and elements to be included in your response, are described in the enclosed Notice of Violation.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

The responses directed by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96.511.

Should you have any questions concerning this letter, please contact us.

Sincerely,

Original signed by
Jerome J. Blake

Caudle A. Julian, Chief
Engineering Branch
Division of Reactor Safety

Enclosures: (See page 2)

Coo1

Enclosures:
1. Notice of Violation
2. NRC Inspection Report

cc w/encls:
C. R. Dietz, Manager
Robinson Nuclear Project Department
H. B. Robinson Steam Electric Plant
P. O. Box 790
Hartsville, SC 29550

J. J. Sheppard, Plant General Manager
H. B. Robinson Steam Electric Plant
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J. D. Kloosterman, Director
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H. B. Robinson Steam
Electric Plant
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bcc w/encls:
Document Control Desk
DRP Section Chief

NRC Resident Inspector
U.S. Nuclear Regulatory Commission
Route 5, Box 413
Hartsville, SC 29550

RII:DRS	RII:DRS	RII:DRP
JLColey	JJBlake	HChristensen
11/15/90	11/16/90	11/19/90

ENCLOSURE 1

NOTICE OF VIOLATION

Carolina Power and Light Company
H. B. Robinson

Docket No. 50-261
License No. DPR-23

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> This is a Severity Level IV violation (Supplement I.D.3).

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FOR THE NUCLEAR REGULATORY COMMISSION

original signed by
Jerome J. Blake

Caudle A. Julian, Chief
Engineering Branch
Division of Reactor Safety

Dated at Atlanta, Georgia
this 20th day of November 1990

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

Report No.: 50-261/90-24

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

Docket No.: 50-261 License No.: DPR-23

Facility Name: H. B. Robinson

Inspection Conducted: October 29 - November 2, 1990

Inspector: J. L. Coley 11-15-90
Date Signed

Approved by: J. J. Blake, Chief 11/16/90
Materials and Processes Section
Engineering Branch
Division of Reactor Safety
Date Signed

SUMMARY

Scope:

This routine, unannounced inspection was conducted in the areas of observation of inservice inspection work activities, review of radiographs, observation of code repair activities for A and C accumulators, implementation of Generic Letter 90-05, and review of licensee actions relating to NRC Compliance Bulletin 87-02 (Technical Instruction 2500/27).

Results:

Inservice inspection efforts at H. B. Robinson have effectively identified two significant areas of material degradation caused by integranular stress corrosion cracking (IGSCC). The two areas were (1) the control rod guide tube support pins and, (2) the upper level transmitter nozzles for A and C accumulators. Automated ultrasonic examinations of the outlet nozzle to reactor vessel weld (RPV-33) revealed an indication which required extensive investigation prior to the final acceptance of the indication. In addition, preliminary ultrasonic data from examinations being conducted on Steam Generator A, Weld 5, (upper girth weld) has also revealed indications that may be indicative of cracks. CP&L's management, engineering, and inspection personnel have responded very effectively to insure that technical issues are resolved in a manner that will insure plant safety. In the areas inspected, one violation was identified 50-261/90-24-01, "Welds not Identified on Inservice Inspection Isometric Sketches" paragraph 4. No deviations were identified.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *R. L. Barnett, Manager, Outage and Modifications
   *W. M. Biggs, Manager, Site Engineering Unit
   *R. L. Crook, Senior Specialist, Regulatory Compliance
   *C. R. Dietz, Site Project Manager, Robinson Nuclear Power Division
   *C. H. Griffin, Senior Engineer, Nuclear Engineering Department (NED)
   *E. M. Harris, Manager, Onsite Nuclear Safety
   *J. D. Kloosterman, Director, Regulatory Compliance
   *G. V. Nuckols, Engineer, Nuclear Engineering Department
   *C. R. Osman, Principle NDE Specialist, Technical Services Department
   *M. F. Page, Manager, Technical Support
   *M. F. Powell, Nuclear Engineering Department
   *R. M. Smith, Manager, Maintenance
   *D. C. Stadler, Onsite Licensing Engineer
   *R. B. Weber, Senior Specialist, Inservice Inspection
   *H. J. Young, Manager, QC/QA

   Other licensee employees contacted during this inspection included engineers, technicians, and office personnel.

   NRC Resident Inspectors

   *L. W. Garner, Senior Resident Inspector
   *K. R. Jury, Resident Inspector

   *Attended exit interview on November 2, 1990

2. Inservice Inspection (73753) Unit 2

   The inspector observed inservice inspection (ISI) work and work activities to ascertain whether examination, repair, and replacement activities associated with ASME Class 1, 2, and 3 components were performed in accordance with Technical Specifications, the applicable section and revision of the ASME Code, correspondence between NRR and the licensee concerning relief requests, and requirements imposed by NRC/industry initiatives. H. B. Robinson is presently in the fifth and last outage of the second inspection interval. The applicable Code for this interval is the ASME B&PV Code, Section XI, 1977 Edition, with addenda through Summer 1978. This inspection was conducted to supplement surveillance activities reported in Inspection Report 50-261/90-21.

a. Volumetric examination of Vessel/Nozzle Welds using the automatic Ultrasonic Technique

The inspector returned to the H. B. Robinson facility on October 29, 1990 and discovered that all automated in-vessel ultrasonic examinations had been completed on October 28, 1990. The vessel/nozzle examinations were performed by Southwest Research Institute (SWRI) examiners utilizing Sonic Mark II ultrasonic instruments. The ultrasonic data was then processed through an enhanced data acquisition system (EDAS) which allowed the data to be recorded and displayed with computer graphics. H. B. Robinson Ultrasonic Procedure "AUT-15" was used to conduct the vessel/nozzle examinations. The ultrasonic examinations had revealed recordable indications in several vessel and nozzle welds. The inspector selected the three welds with indications to evaluate the SWRI examination and evaluation results. The three welds selected are listed below:

Weld No.	Examination No.	Weld Description
RPV-29	79	Outlet Nozzle to Shell at ten degrees
RPV-33	87	Outlet Nozzle to Shell at 250 degrees
RPV-4	29	Intermediate shell to Upper Shell longitudinal Weld

In addition to the above, the inspector reviewed certification records for all SWRI examiners and equipment on site.

Review of the data for welds No. RPV-29 and RPV-4 revealed that the indications recorded in these welds were acceptable and indicative of small slag inclusions which resulted during fabrication of the vessel. However, an indication was detected during the examination of nozzle to shell weld, RPV-33, which was conducted with 0 degree longitudinal and 45 degree shear wave search units located in the nozzle bore. The indication was detected with the 45 degree search unit and appeared to be located in the weld material at an azimuth of 237 degrees. A subsequent re-look of the area containing the reflector was performed using the 45 degree search unit that initially detected the indication. This re-look confirmed the existence of the indication.

This initial re-look showed at least two individual reflectors which when combined, constituted an indication that appeared to be rejectable relative to the Section XI, 1977 with Addenda through the Summer of 1978, Table IWB-3512, acceptance standards.

Subsequent re-examinations were performed by conducting three separate but identical calibrations using the same 45 degree search unit used to initially detect the indication. Two different instruments were used in order to assure that an instrumentation problem did not exist. Three separate sets of data were collected and correlated to provide the final sizing data package. These data sets all showed the indication to be significantly lower in amplitude than the original re-look and acceptable in accordance with Section XI criteria. Additionally a focused 45 degree, 2.25 mHz, shear-wave, search unit, a 35 degree, longitudinal-wave, 2.25 mHz, tip-diffraction, search unit and a 0 degree, 2.25 mHz, tip-diffraction, search unit were used to scan the indication area. The indication could not be detected with either of the tip diffraction units or with the 45 degree, 2.25 mHz, focused unit.

The conclusion drawn from this data package is that the indication is of an acceptable size in accordance with the criteria of Section XI, 1977 Edition with Addenda through the Summer of 78, Table IWB-3512. The apparent cause (supported by calibration data) of the amplitude difference between the initial re-look examination and the subsequent re-examinations was due to a poor cable connection. Fabrication radiographs were also reviewed by the inspector for weld RPV-33. These radiographs revealed several small (acceptable) slag inclusions in this weld. However, the orientation of these radiographic indications with the indication detected during the examination of the nozzle could not be confirmed.

b. Ultrasonic examination of Steam Generator A, Weld 5 (Upper Girth Weld)

The inspector also observed Westinghouse examiners conducting manual ultrasonic examinations on Steam Generator A, Weld 5. These were licensee augmented examinations, since the ASME Code required examinations for this inspection interval had been satisfied with the ultrasonic examination of weld 5 for Steam Generator B (see Region II Inspection Report 50-261/90-21 for details).

The licensee's purpose for expanding their ultrasonic examination of weld 5 to include Steam Generators A and C was to determine whether any of H. B. Robinson steam generators were cracked. Significant Cracking adjacent to this weld has been detected and repaired at a number of sites including Indian Point, Surry, and Zion. Although, the licensee ultrasonic examination of Steam Generator B recorded some isolated indications, these indications were not considered indicative of cracks. However, preliminary in-process examination of Steam Generator A, weld 5, has recorded indications which have been tentatively plotted. These plots indicate that the ultrasonic reflectors are at the vessel ID and that they run in a circumferential direction ranging from the outer edge of the weld to one inch beyond the weld edge in the base material. All of the indications were of relatively low amplitude thus far. However, these indications are located where cracks have been confirmed at other utilities. Examination, sizing and evaluation efforts are severely restricted for the H. B. Robinson steam generators due to a shield wall constructed around each steam generator. Therefore, during the inspector's exit meeting these restricted conditions were presented to the plant management, along with NRC's recommendation that the secondary side of the steam generator be opened, and weld 5 be examined using the magnetic particle examination method. No commitment to open the steam generator was made by plant management during the exit meeting, nor was one expected at that time.

c. Repair and Replacement Activities

(1) Repair Activities

While performing a hydrostatic test on the Safety Injection (SI) Accumulator "C", a leak was detected in the stainless steel nozzle coupling for one of the upper level transmitter lines (2-inch diameter, 2000# half-coupling, pipe line number 2-SI-601R). There are three such SI Accumulators at the Robinson Nuclear Plant (RNP).

The RNP SI accumulators were manufactured for Westinghouse by the Delta Southern Corporation, formerly located in Baton Rouge, Louisiana. Delta Southern Corporation is no longer is business.

A search by the licensee of the Nuclear Plant Reliability Data System (NPRDS) revealed that the Prairie Island Plant (Northern States Utilities) had previously experienced a similar leak in one of their Delta Southern Corporation accumulators. Their nozzle failure was also in a two-inch diameter level transmitter nozzle. Westinghouse had performed a failure analysis of the Prairie Island nozzle and determined the cause of failure to have been intergranular stress corrosion cracking (IGSCC). The stresses responsible for initiating the IGSCC in the Prairie Island nozzle were believed to have been the result of an improperly fit-up and welded socket weld (pipe apparently bottomed into the socket prior to welding).

Failure analysis testing on the H. B. Robinson (HBR) failed nozzle was performed by the CP&L Metallurgical Services Section located at the Harris Engineering and Evaluation (E&E) Center (New Hill, N.C.). The failure analysis results for the HBR nozzle indicated the cause of the leak to be IGSCC. It should be noted that for IGSCC to initiate, the following three conditions must exist simultaneously: tensile stresses, sensitized base material, and a corrosive environment.

The axially oriented crack had initiated on the ID of the coupling. Although the crack appeared to extend along most of the coupling ID length which was contained within the tank shell, that portion of the crack that was open to the coupling OD was only approximately 3/16 inch long and did not appear to extend into the nozzle attachment fillet weld reinforcement. Unlike the Prairie Island nozzle, socket joint fit-up and welding was not a contributing factor in the failure of the HBR level transmitter nozzle coupling. CP&L believes that the full penetration weld (double-V groove design), connecting the nozzle to the shell, is the source of residual stresses which initiated the cracking. The axial orientation of the cracking seems to provide some credence of this hypothesis.

The failed nozzle coupling was sensitized. The sensitization of the coupling was attributed to a combination of the welding to the coupling and the post weld heat treatment (PWHT) performed by the manufacturer after the completion of the welding to the accumulator. Typically, when a manufacturer plans to perform PWHT on a component which contains stainless steel materials such as the 304/316 grades, a low carbon grade of the material containing .035 percent maximum carbon content (e.g., Types 304L or 316L) would be recommended. These low carbon grades of stainless steel are much less susceptible to sensitization and are therefore typically considered immune to IGSCC. The carbon content for the failed coupling was determined to be .062 percent carbon during the failure analysis. No certified material test reports were found for the 2, 1, and 3/4 inch nozzles although, the Manufacturers' Data Reports indicated that the nozzle couplings are SA-182-F-304 ELC (extra low carbon). Although the presence of chlorides and sulfates was confirmed by water leachable analyses of the failed coupling, the exact corrosive environment which contributed to the crack initiation was not determined.

The licensee subsequently performed liquid penetrant and ultrasonic examinations on all eight nozzles for each accumulator. These examinations revealed that accumulator A also had a two inch nozzle with a crack indicative of IGSCC and accumulator B had a forging defect in a one inch nozzle that was apparently caused during fabrication. The indication extended into the coupling for approximately 1/2 the coupling through-wall and may run the entire length of the coupling but did not extend to the ID surface of the pipe.

The inspector reviewed drawings, material certification reports, engineering evaluations of the failure, and engineering evaluations for the repair of the failed nozzles, Structural Integrity Associates repair program for the nozzles and ultrasonic and liquid penetrant examination reports. In addition, cognizant personnel including metallurgist, welding engineers, system engineers and inspection personnel were interviewed to assess the failure, methods of repair, expanded inspection results, determine whether other components were involved, and to determine if augmented inspections would be performed in the future on nozzles that are susceptible to failure. The inspector also observed the nozzle excavation areas to determine whether they met applicable code repair criteria for weld repairs without PWHT.

The inspectors review of the above revealed that the licensee was performing the technical assessment and repair activities in accordance with plant procedures and the ASME Code.

(2) Replacement Activities

Occurrences of failed reactor control rod guide tube support pins (split pins) during the early 80's caused Westinghouse to issue a service letter identifying plants with potential for failures (suspect heats identified). Robinson could not identify heats associated with their pins at the time but assumed that they were satisfactory since no problems had been encountered. During March 1989 operations, a split pin failed at H. B. Robinson due to cracking. A part of the failed pin was injected into the "C" Steam Generator and its retrieval required a forced shutdown on April 3, 1989. As a result, Robinson committed to ultrasonic examination of all split pins during the current outage. Ultrasonic examinations were conducted by Westinghouse with results as follows:

38 pins with indications indicative of crack (represents 36 percent of the 106 total pins)
- 37 pins of the above revealed cracks in the pin collar to shank change of section area
- 2 pins (including one of the 37) have cracks in the leaf area

As a result of the high failure rate delineated above, the licensee has decided to replace all 106 spilt pins this outage with pins that are less susceptible to crack. The inspector reviewed the Westinghouse ultrasonic procedure used to detect the cracks and discussed the inspection findings with cognizant engineers.

(3) Modifications (57090)

In addition to the repair and replacement activities described above the inspector audited welding activities by reviewing completed radiographs. The radiographs reviewed were for welds on an electrical penetration modification. ASME, Section III, Subsection NE, Class MC was the applicable Code for this review. Radiographs for the following welds were reviewed:

Weld ID	Diameter	Drawing No.
C-5-1 (Cut 1, Repair 2)	10 inches	Conax 7EEG-6000
E-1-1	10 inches	Conax 7EEG-6000
E-10-1 (Cut 2, Repair 3)	10 inches	Conax 7EEG-6000

d. Review of Licensee's Implementation of NRC Generic Letter (GL) 90-05

On June 15, 1990, GL 90-05 was issued to all holders of operating licenses for nuclear power plants. This GL provided guidance for performing temporary non-code repairs of ASME Class 1, 2 and 3 piping. A summary of the guidance is as follows:

- NRC approved relief requests are required for all temporary non-code repairs in Classes 1, 2 and 3 piping that is impractical to repair during power operations.

- Flaws detected during scheduled shutdowns must be Code repaired prior to restart.

- Temporary repairs can remain until the next scheduled outage exceeding 30 days, but no later than the next scheduled refueling outage.

- Temporary repairs in Class 1 and 2 piping and high energy Class 3 piping must meet certain load-bearing requirements similar to that provided by engineered weld overlays or engineered mechanical clamps.

- Temporary repairs in other Class 3 piping may be analyzed using the NRC specified "wall-thinning" or "through-wall flaw" methods.

The inspector discussed GL 90-05 with CP&L's Outage and Modification Manager and reviewed CP&L's Plant Program Procedure PL-025, "In-Service Inspection Program," and Technical Support Management Manual TMM-015, "Inservice Inspection Repair and Replacement Program" to ensure that necessary requirements have been properly established to implement Generic Letter 90-05 Guidance. During discussions with the Manager of Outage and Modifications the inspector was assured that Robinson would have no non-code repairs on any ASME Code Class components when Unit 2 resumes power operations.

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

3. NRC Compliance Bulletins (Temporary Instruction 2500/27)

(Closed) NRC Compliance Bulletin 87-02, Fastener Testing to Determine Conformance with Applicable Material Specifications. The purpose of this bulletin was to request that licensee's 1) review their receipt inspection requirements and internal controls for fasteners and 2) independently determine, through testing, whether fasteners (studs, bolts, cup screws and nuts) in stores at their facilities met required mechanical and chemical specification requirements. On January 22, 1988 CP&L Letter Serial No. NLS-88-013 was submitted to Region II. This letter provided the results of CP&L's review and testing program. These results indicated that 52 of the 54 fasteners tested met the applicable ASTM specifications. The two fasteners not in complete compliance with their specifications were evaluated and determined to be acceptable for their intended use. CP&L's review concluded that their current procedure for receipt inspection and material handling met or exceeded applicable requirements. On May 22, 1989 NRC issued Temporary Instruction (TI) 2500/27. This instruction was issued to provide inspector guidance in evaluating the adequacy of certain licensee's root cause analysis and the implementation of corrective actions in response to NRC Bulletin 87-02. The instruction listed CP&L's Robinson plant as having one fastener that was significantly out of specification. The fastener listed was Sample RNP-006, a 3/4" diameter ASME SA 193 GRB8 Capscrew.

The sample capscrew met the testing requirements with the exception of hardness. The small difference between the measured valve of 26 Rockwell C scale (equivalent to 102 Rockwell B scale) and the specification maximum of 100 Rockwell B scale was not considered by the licensee to be significant. The inspector reviewed the licensee's response to Bulletin 87-02 and CP&L's backup data file that supported their response. In addition, discussions were held with metallurgist working for the licensee and NRC concerning the two points difference in Rockwell B scale hardness. It was the consensus opinion of all the specialists that the two points difference on the Rockwell B scale did not indicate a significant difference in hardness. Therefore, the inspector concluded that the response to NRC Compliance Bulletin was adequate and no further action is required.

4. Action on Previous Inspection Findings (92701, 92702)

(Closed) Unresolved Item 50-261/90-21-01, Welds not Identified on Inservice Inspection Isometric Sketches." This item reported that isometric sketches used in the ISI program to identify the configuration of piping, weld joint locating and weld population in the ISI program were found to be in error in that welds were missing from the sketches. The unresolved item was open until the inspector could discuss this discrepancy with the Office of Nuclear Reactor Regulations (NRR) in order to determine the affect this finding would have on the ISI program currently approved by NRR and also on the ISI program that has been prepared by the licensee for the 3rd inspection interval which will start next refueling outage. Subsequent discussions with NRR concluded that this discrepancy could be handled adequately with 10 CFR 50, Appendix B, Criterion V, violation.

Therefore, this unresolved item is considered closed and the issue upgraded to a violation (50-261/90-24-01, "Welds not Identified on Inservice Inspection Isometric Sketches").

5. Exit Interview

The inspection scope and results were summarized on November 2, 1990, with those persons indicated in paragraph 1. The inspector described the areas inspected and discussed in detail the inspection results listed below. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

(Open) Severity Level 4, Violation 50-261/90-24-01, "Welds not Identified on Inservice Inspection Isometric Sketches" Paragraph 4.