U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION IV

(This Report Contains Investigation
Information See Paragraph 14)

Report No. STN 50-482/80-13

Docket No. STN 50-482 Category A2

Licensee: Kansas Gas & Electric Company
Post Office Box 208
Wichita, Kansas 67201

Facility Name: Wolf Creek, Unit No. 1

Inspection at: Burlington, Coffey County, Kansas

Inspection Conducted: July 1980

Inspectors: C. R. Oberg 9/3/80
C. R. Oberg, Resident Reactor Inspector (Acting) Date
Projects Section (Paragraphs 1-3, 7-14 & 18-21)

for C. R. Oberg 9/3/80
W. G. Hubacek, Resident Reactor Inspector (Acting) Date
Projects Section (Paragraphs 1-6, 15-17, 20 & 21)

Approved: W. A. Crossman 9/4/80
W. A. Crossman, Chief, Projects Section Date

Inspection Summary:

Inspection during July 1980 (Report No. STN 50-482/80-13)
Areas Inspected: Routine, announced inspection by the Resident Reactor Inspectors (RRI) (Acting), including follow up to previous inspection findings; safety-related concrete placements; safety-related pipe welding; QC personnel certification; construction deficiencies; investigation of an allegation relating to Essential Service Water (ESW) Pipe; deficiency reports and nonconformance reports; review of records relating to reactor vessel installation; storage of mechanical core structures; and ESW system records and site tours of current construction activity. The inspection involved one hundred twenty-six inspector-hours on site by two NRC inspectors.
Results: No items of noncompliance or deviations were identified.

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DETAILS

1. **Persons Contacted**

   Principal Licensee Employees

   M. E. Clark, Manager, Quality Assurance, Site
   D. W. Prigel, Assistant QA Manager, Site
   G. L. Fouts, Construction Manager
   S. D. Bostom, QA Auditor
   L. Borders, Construction Engineer
   R. M. Stambaugh, QA Engineer
   D. G. Plasce, QA Engineer
   J. A. Cherry, Contract Administrator
   J. L. Stokes, Construction Engineer
   T. H. Young, QA Engineer
   N. W. Hottel, Electrical Site Liaison
   R. Brown, SNUPPS, Site Representative

   Daniel International Corporation

   W. E. Hitt, Project Manager
   V. J. Turner, QA Manager
   D. L. Jones, QC Manager
   R. D. Scott, Construction Manager
   C. L. Phillips, Engineering Manager
   N. Criss, Audit Response Coordinator
   M. Pfeifer, Quality Control, Civil
   F. Cherry, Concrete Superintendent
   C. Griffin, Concrete Superintendent (Special)
   R. Stout, QC Inspector
   J. Ferguson, QC Services Engineer
   D. J. Dennison, Assistant QC Manager
   J. M. Ayres, QA Engineer
   J. T. Goodwin, Electrical Engineer
   R. Schofield, Assistant P. W. Engineer
   T. Newman, Mechanical Engineering

   The RRIs (Acting) also talked with and interviewed other licensee employees and contractor personnel including members of the QA/QC and engineering staffs.

2. **Licensee Action on Previous Inspection Findings**

   (Closed) Unresolved Item (STN 50-482/79-19): Control of Vendor Manuals. The RRI (Acting) verified that Daniel Quality Assurance Audit Findings (QAAF) Nos. 47-05 (July 19, 1978) and 47-06 (July 19, 1978) have been corrected. The corrective actions were documented in Daniel Quality Assurance Report (QAR) Nos. 50 (November 5, 1979) and 55 (April 2, 1980). The Document Control Section now receives a "Supplier Print Control

Register" for Instruction Manuals. Daniel Construction Procedure AP-IX-03, "Document Control," Rev. 9 (April 14, 1980), is currently under revision (Rev. 10). The RRI (Acting) reviewed the draft procedure and determined that adequate controls for manuals now exist. Additional filing space has been provided in the Document Vault.

This item is considered closed.

(Closed) Unresolved Item (STN 50-482/79-07): Storage of Reactor Vessel and Reactor Pressure Vessel Head. This specific activity was inspected in March 1980, and resulted in an item of noncompliance. (See Inspection Report STN 50-482/80-05).

Inasmuch as this item was upgraded to an item of noncompliance, this unresolved item is thus considered closed and is documented here for record purposes.

(Closed) Unresolved Item (STN 50-482/79-07, paragraph 6.b): Replacement Splices. The RRI (Acting) reviewed the revision to WP-IV-102, paragraph 3.9, which specifies that the Civil Engineer is to be informed of failing transition Cadweld splices and is to direct replacement requirements to the Cadweld Foreman. The RRI (Acting) had no further questions on this matter.

This item is considered closed.

(Closed) Unresolved Item (STN 50-482/79-09): Review of Reactor Vessel Internals Storage Procedures. RMI-W-042 (Q), Rev. 4 now contains specific frequencies for maintenance inspection requirements. The discipline engineer is required to establish the "maintenance requirements" for the equipment to be inspected in accordance with Daniel QCP-I-01. This includes frequency of performing maintenance checks.

This item is considered closed.

(Open) Infraction (STN 50-482/79-09): Failure to Follow Storage Procedures for Reactor Vessel Internals. The RRI (Acting) determined that deficiencies reported in Deficiency Report (DR) 1SD1657M were corrected and dispositioned by Bechtel on June 6, 1979.

DR 1SD1471M has not yet been dispositioned. This had been forwarded to the Westinghouse site representative for corrective action. No action has as yet been taken to resolve the problem. This is considered unresolved and will be reviewed in a subsequent inspection. All other commitments made in KG&E letter of July 6, 1979, have been completed. The RRI (Acting) reviewed documents and held discussions with cognizant personnel to verify the completed action.

This item is held open only for resolution of 1SD1471M. This matter was discussed with the licensee.

3. **Site Tours**

The RRI (Acting) made daily tours of the site to observe construction activities and inspect housekeeping and equipment storage. Some minor discrepancies in equipment storage were observed but were promptly corrected when pointed out to the licensee. The RRI (Acting) also toured the UHS Dam and the ESW System.

No items of noncompliance or deviations were identified.

4. **Inspection of Concrete Placement**

The RRI (Acting) observed activities related to placement of concrete for pour No. OC281W06 in the Reactor Building dome. The placement was made during the night of July 20-21, 1980, to avoid the effects of the intense daytime heat upon the placing crews and the concrete. Approximately 292 cubic yards of Mix 66-E1A-N were placed.

Some early problems were experienced with deposition and consolidation of the concrete in the first lift; however, the problems were identified and corrective action was taken by craft supervisors and QC inspectors and it appeared that adequate consolidation was achieved. The RRI (Acting) also observed that lighting provided was marginal, particularly during placement of the first lift. Licensee representatives stated that action would be taken to prevent recurrence of these problems during the next dome placement.

The RRI (Acting) reviewed the following documents related to placement OC281W06:

"Concrete Placement Checklist"

"Concrete Placement Card" with attached special instructions

"Preplacement Checklist"

"Tendon Sheathing and Embedded Anchorage Checklist"

"Post Placement Checklist"

The RRI (Acting) also reviewed the following documents related to placement OC281W04 which was completed on June 25, 1980:

"Batch Tickets" No. 20677 through 20715 indicating 360 cubic yards of Mix 66-E1A-N were placed

"Batch Adjustment Forms"

"Total Moisture Content of Aggregate by Drying"

"Compressive Strength of Cylindrical Specimen" for seven day breaks

"Concrete Placement History Checklist"

"Concrete Placement Checklist"

"Concrete Placement Card"

"Preplacement Checklist"

"Tendon Sheathing and Embedded Anchorages Checklist"

"Post Placement Inspection Report"

No items of noncompliance or deviations were identified.

5. Observation of Pipe Welding Activities

The RRI (Acting) observed activities related to welding of joint FW 300 in the three inch stainless steel pipe attached to the bottom of the boric acid tank. Requirements for the weld were contained in Engineering Change Request I-M-03 BG 16 (Q) ECR-01 and the attached Weld Record Card (WRC) which referenced Technique Sheet N-8-8-B-2, Revision 4 of Procedure CWP-507, "Welding of Stainless Steel."

The RRI (Acting) observed prewelding inspection performed by a Level II mechanical/welding QC inspector and a trainee. All items listed on the WRC were inspected, including fitup, cleanliness, purge and shield gas flow rate, oxygen content of purge, weld filler metal control, welder identification and certification, preheat temperature, and initial welding current. The RRI (Acting) observed welding of the root pass.

The RRI (Acting) also observed welding of the root pass and part of the subsequent passes for Field Weld F081-003 in the ESW pipe located in the pit west of the Control Building. Requirements for this weld contained in the WRC appeared to be met.

No items of noncompliance or deviations were identified.

6. Certification of QC Inspectors

The RRI (Acting) reviewed documentation pertaining to certification of nine QC inspectors contacted during the inspection period to determine if they were certified in accordance with the requirements of Construction Procedure AP-V1-01, "Indoctrination, Training, and Qualification of Quality Control Personnel."

The RRI (Acting) observed that the required documentation was available for all of the nine inspectors and that certification requirements appeared to have been met; however, the RRI (Acting) noted that there were inconsistencies in a large percentage of the "Documents of Certification" in the QC inspector's files in that the dates of qualification were chronologically earlier than the approval signatures by the Project QC Manager and the individuals who recommended that the QC inspectors be certified. The reasons for these discrepancies could not be determined since the individual responsible for preparation of the "Documents of Certification" was on leave.

This matter is considered unresolved and will be followed during a future inspection.

7. Potential Construction Deficiency - Hold Down Bolts for Essential Service Water (ESW) Pumps

The RRI (Acting) reviewed the Bechtel evaluation reports submitted to KG&E on May 23, 1979, on the ESW pump hold down bolts. Bechtel calculations showed that ". . . the actual maximum nozzle and seismic forces are less than one-fifth of the allowable values. Therefore, the presently specified anchor system has a factor safety of more than five."

The licensee reported to the NRC (Region IV) on May 27, 1979, that this item was no longer considered or to be a reportable item. Based on the information reviewed, the RRI (Acting) concluded that this is not a reportable construction deficiency.

8. Potential Construction Deficiency - Incorrect Color Coding of Class 2 & 3 Bolts

Stud bolts and nuts, which were purchased as ASME Section III Class 1, 2 & 3, were incorrectly color coded for class designation. This was reported to the NRC on December 19, 1979. An investigation of the circumstances made by Daniel was reviewed by the RRI (Acting). On-site QC inspectors have examined all bolts and nuts in stock and corrected color coding of those that could be traced back to their documentation. Those bolts and nuts which were issued to the field were traced and discrepancies were corrected. Only five connections were found bolted and all five had the correct class of bolts for the class of connection. The investigation shows that none of the incorrectly marked bolts were used in a location requiring Class 1 bolts.

The RRI (Acting) concluded that the event is not reportable under 10 CFR 50.55(e).

9. <u>Potential Construction Deficiency - Westinghouse 7300 Series Process Control System - Part 21 Report</u>

Westinghouse letter to OIE (NS-TMA-2124), dated August 23, 1979, reported a potential substantial safety hazard under the provisions of 10 CFR Part 21. This report related two technical problems in the 7300 Series Process Control System I and C cards in units of the type currently in storage at the Wolf Creek site. A visit was made by Westinghouse engineers to Wolf Creek on April 7, 1980, to inspect the affected I and C cabinets. Four discrepancies were identified; however, they were not related to specific safety functions. The RRI (Acting) reviewed the documentation available on site regarding the failed cards and component identified in the Westinghouse report. Nonconformance Report 1SN1924ER was generated to document the requirements for replacement parts (Printed Circuit Boards).

The RRI (Acting) concluded that this event is not reportable under 10 CFR 50.55(e).

10. <u>Potential Construction Deficiency - Concrete Defects in Reactor Cavity (See Inspection Reports STN 50-482/80-07 and 80-09)</u>

Two instances of damaged concrete were identified in the refueling pool floor at elevation 2021'7" adjacent to the reactor cavity. These were separate occurrences and did not have a common origin. Both instances of damaged/cracked concrete were documented and evaluated by Bechtel, the lead A/E for Wolf Creek. Bechtel's evaluation ". . . indicated that neither of the two instances of damaged concrete, if left uncorrected, would have become a significant safety problem throughout the operating life of the Wolf Creek No. 1 Unit . . . ." [1] The following is a summary of the causes and disposition:

a. <u>Cracks Resulting From Excessive Strain on Seal Ring Plate Lugs</u>

The cause of the deficiency was discussed in Inspection Report No. STN 50-482/80-09. No structural damage to the plate was noted. The subject liner plate is non-Q and damage to the concrete was not significant. The item is therefore not reportable under 10 CFR 50.55(e).

b. <u>Spalled Concrete in Reactor Cavity</u>

The cause of the spalled concrete was determined to be excessive forces exerted on the liner plate during the adjustment and fitting up process. These forces were transmitted to the bottom of the liner plate and the W4 beams which in turn caused the cracking of the concrete. The unsound concrete was limited to the immediate vicinity of the ledge around the reactor cavity and had no effect on the ability of the structure to perform its safety-related function. Unsound concrete was removed and all damage was identified and documented. The item was therefore considered not reportable under 10 CFR 50.55(e).

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[1] KG&E letter, Koester to RIV, dated June 23, 1980

The RRI (Acting) reviewed the disposition NCRs 1SN1808C and 1SN2015C and the Bechtel evaluation. The causes of the damage were identified and an adequate and conservative discussion of the factors involved was included in the evaluation. The site of the damage was inspected by the RRI (Acting). Repairs were accomplished in accordance with the Bechtel dispositions.

11. Potential Construction Deficiency - Incorrect Concrete Mix Used in Auxiliary Building Roof

A potential construction deficiency was reported to RIV on May 28, 1980. This deficiency involved the use of the wrong concrete design mix. The correct mix had a specified strength of 5000 psi at 90 days. The cylinders of the actual mix used were tested and found to be over 6000 psi at 28 days. The average strength was 6470 psi at 28 days. The RRI (Acting) concluded that the matter is not reportable under 10 CFR 50.55(e).

12. Potential Construction Deficiency Report - Nonconforming Threads on 3¼ Inch Diameter Studs and Nuts for NSSS Steam Generator Lateral Supports

On June 6, 1980, a potential construction deficiency was reported to RIV relating to the apparent failure of several nuts at Wolf Creek to meet the dimensional standards of ASA Standard B1.1-1960. NCR report No. 1SN1548C was dispositioned by Bechtel to replace the nonconforming nuts with those that meet the ASA standards. The studs and nuts were manufactured by Southern Bolt Company, Shreveport, Louisiana. When the nonconformance was identified, twenty-four studs and nuts were embedded in concrete at Wolf Creek. The same problem was identified at the Callaway site. An analysis of the problem was done by Bechtel which was reviewed by the RRI (Acting). The results of the investigation and testing program by Bechtel indicate a high level of confidence exists that all stud-nut combinations at the Wolf Creek jobsite can adequately support the maximum design load. The 912 kip minimum expected thread shear capacity provides a factor of 2.75 against a thread shear failure at the maximum design load of 334 kips. The report concluded that had the nonconformance gone undetected there would have been no impact on the health and safety of the public. The RRI (Acting) agreed with this conclusion. This item is considered closed.

13. Potential Construction Deficiency Report - Linear Indications in Pipe Spool for Essential Service Water System

A potential construction deficiency was reported to Region IV on May 8, 1980. On June 6, 1980, a report was submitted to RIV on the ESW system indicating that the item is not reportable under 10 CFR 50.55(e). The RRI (Acting) examined the pipe spool and reviewed the records relating to the reported linear indications. As stated, removal of the indication results in a wall thickness less than the minimum specified (.365"). Approved disposition of the DR (No. 1SD3603M) indicates the pipe will be replaced.

The RRI (Acting) thus concluded that the item is not reportable under 10 CFR 50.55(e) based on this isolated occurrence. Further, the pipe spool is in the return line of the ESW system. Any type of failure of this pipe spool due to pipe flaws could not cause a significant reduction in safety. For record purposes this item is considered closed.

14. Investigation of Essential Service Water Pipe Internal Cleanliness

NRC Investigation Report No. STN 50-482/80-06, Allegation No. 12, refers to the Essential Service Water (ESW) system and an allegation that mud and rags were left inside a previously installed segment of pipe south of the ESW valve house. The findings of the investigation were that the item could not be substantiated in that access due to safety reasons was not possible at that time (March 18-24, 1980).

On July 9, 1980, the RRI (Acting) witnessed an entry into one segment of the supply pipe. A volunteer pipefitter entered the 30" pipe on a dolly for a distance of approximately 450 ft. The pipe was found to be free of foreign objects (rags, mud, etc.,). Four to six inches of water in the pipe prevented further entry.

On July 18, 1980, RIV received a telephone call from KG&E who reported the following information relating to this allegation:

a. A forman* sent two men, one at a time, into the ESW pipe about the time of the NRC investigation (STN 50-482/80-06) when he became aware of the allegation.

b. Rags were found and brought out by both men.

c. The foreman* reportedly acted without authority, risking the lives of the men. There was water present in the pipe.

d. The incident was not reported to the proper authorities.

e. The rags were left in the pipe as a "water dam" to facilitate welding.

The RRI (Acting) proceeded to investigate this matter through review of records and interviews with personnel and determined the following information:

a. A general foreman had "requested" two men to enter the ESW pipe and remove the rags. This apparently occurred during the time of the investigation. The general foreman was terminated on July 16, 1980, for "gross violation of Safety Rules." He had not reported his actions to his supervisors.

*It was determined later that a "general foreman" was involved.

b. Discussions with the two men who entered the pipe confirmed that rags had been present. The two return lines were entered. The rags were used as a "dam" to prevent the water from draining down into a weld area. After the last segment of pipe was welded in place, the rags could not be removed.

c. No other foreign material was known to have been left in the pipe. The water was from a hydro test done on the pipe runs earlier in construction.

d. None of the men involved in this incident had been interviewed by the investigators in reference to report No. STN 50-482/80-06.

The RRI (Acting) concluded that, based on the information obtained, the allegation was confirmed but had no merit. It must be noted that the rags were in the "return" portion of the ESW system. During the ESW system "flush," any and all foreign material would have been discharged to the cooling lake area.

15. <u>Review of Nonconformance Reports (NCRs) and Deficiency Reports (DRs)</u>

The NCRs and DRs listed below were reviewed in order to determine conformance to requirements for reporting construction deficiencies:

1SN1588C	1SD4287M
1SN1600C	1SD4094M
1SN1644C	1SD4057C
1SN1723C	1SD3332M
1SN1722C	1SD3603M
1SN1773C	1SD2978M
1SN1765C	1SD3515M
1SN1860M	1SD2256M
1SN1874C	1SD3022M
1SN1882M	
1SN1963C	
1SN1504M	

No items of noncompliance or deviations were identified.

16. Installation of Reactor Vessel - Review of Records

The RRI (Acting) reviewed the records of the final setting of the reactor vessel. The following documents were reviewed:

KG&E Surveillance Report No. S-206

NCR 1SN1954M

NCR 1SN1503M (with attachments)

NCR 1SN1644C

NCR 1SN1882M

Westinghouse Print No. 1459F01 (reactor vessel supports)

Discussions were held with Bechtel and Daniel personnel involved with the reactor vessel setting. Westinghouse Mechanical Service Manual, Volume I, Section IV, Page IV-2-3 and AWS D1.1-75, paragraph 3.5.1.9 require 75% surface contact (< 0.010 inch). The following was determined by measurement of the contact between the reactor vessel foot and the wear plate:

Nozzle	Contact
Outlet D	81.17%
Inlet F	74.55%
Outlet H	76.62%
Inlet B	89.46%

It was subsequently determined by Daniel Engineering that Inlet F had in excess of 75% contact (NCR 1SN1954M).

No items of noncompliance or deviations were identified.

17. Storage Requirements for Mechanical Core Structures

The RRI (Acting) reviewed the Receiving and Maintenance Instructions (RMI) W-042 (Q), Rev. 4 for the mechanical core structures (Upper and Lower Internals). Rev. 4 of RMI-W-042 (Q) was signed July 21, 1980, and requires that the "Lower Internals Package" be stored horizontally. By direct observation, the Lower Internals Package is stored (on July 28, 1980) in the refueling canal of the Containment Building in a vertical position. A new RMI (Rev. 5) was issued on July 29, 1980, which specified the actual position. When this item was discussed during the exit meeting on August 7, 1980, the Daniel representative stated that DIC did not have responsibility for storage inspections of the reactor internals. The item is therefore considered unresolved and will be reviewed during a subsequent inspection.

18. Work Hold Agreement

The licensee issued a "Work Hold Agreement" (No. 11) on July 25, 1980, which stopped work on field coatings and surfaces for all safety-related structures. The reason for the work stoppage is that the current construction procedures do not reflect the requirements of Bechtel Specification 10466-A-125(Q), "Technical Specification for Surface Preparation, Furnishing and Application of Special Field Coatings and Surfaces for SNUPPS."

This item will be reviewed in subsequent inspections.

19. Essential Service Water System (ESW)

The RRI (Acting) made a visual inspection of available portions of the ESW piping and valve components. A review was also made of specific welding records and other documents as listed below:

Welding Technique Sheet, N-1-1-BA-1, Rev. 5

NCR 1SN1417M, November 20, 1979, (Defective Spool S011 ESW line)

ECR ICK-205-ECR-05, March 11, 1980

F101 Weld Control Record, Weld F301, March 21, 1980

F101 Weld Control Record, Weld F302, March 21, 1980

Weld Repair Instructions, Weld Z-F301-R1, March 25, 1980

F101 Weld Control Record, Weld F301-R1, March 25, 1980

1SD Deficiency Report, 1SD3603M, March 26, 1980,
Spool 205-S011-ESW Line

NDE Report, PBT-MT-753, March 26, 1980 (Accept)

NDE Report, PBT-MT-754, March 26, 1980 (Reject)

Hydro Test Report, ESW Line, April 7, 1980 (Satisfactory)

NDE Report, PBT-MT-772, March 10, 1980, (Reject)
ICK-205-S011

NDE Report, PBT-UT-257, April 11, 1980, (Reject)
ICK 205-S011 (Minimum Wall Thickness was Exceeded)

"Rapid Memo" (Kleikege to Schofield) Replace 5011, April 22, 1980

Bechtel Memo 108 81-M-201C (18D-3003M), April 30, 1980

ECR ICK-205-ECR-05 (Change 2), May 28, 1980

DIC Inter-Office Communication re: Quality Performance, August 4, 1980

ICP 202, Rev. 0, to Procedure AP-VI-02, Rev. 8

During the record review, specific attention was made of a series of comments inserted by QC personnel in the Weld Control Record, Weld F301, Drawing No. TC-K205-ECR05. The comments concerned the acceptability of the 30" pipe weld joint fit up. After discussion with Daniel QC personnel and visual examination of the weld, several facts were determined. These are described below:

a. The weld (F-301) offset was initially determined to be excessive (5/32") and documented by the QC inspector (Level II) to be incorrect and as exceeding code limitations. The welding foreman protested to the QC inspector's supervisor.

b. After a discussion between the QC inspector and his supervisor, the weld preparation was signed off. The next day, the QC inspector changed his mind and stated it was incorrect. After further discussions, the supervisor approved further welding.

c. Further discussions between the Assistant Manager, Quality Control and QC inspector developed. The weld (the root pass was now complete) was ground out in the area of question and the offset again measured by a Hi-Lo gauge. It was found to be and recorded as 3/32". The weld was then repaired and subsequently passed NDE.

The basis for acceptance of pipe fitup is found in the Daniel Welding Technique Sheet N-1-1-BA-1 and ND-4000, paragraphs ND-4232 and ND-4233 of ASME Section III, Division 1. Due to problems in alignment, a large tack weld (approximately 14 inches) had been placed in the area of concern to the QC inspector. Based on an external measurement (Cambridge gauge) in the center of the tack weld, the offset was thought to be 5/32". During the inspection process related above, the deficiency report which had been filled out by the QC inspector was not located. It was then determined that since the report had no basis, as determined by QC supervisor, it was not assigned a report number and therefore not entered into the DR/NCR log. A copy of the original DR had been retained by the QC supervisor and was provided to the RRI (Acting).

The procedure for handling DRs was discussed with the QC Manager. As a result, a change to the procedure for the processing of DRs was made. The change, ICP-202, Rev. 0 requires that all DRs initiated by a QC inspector be reviewed and in part states that, if "At any time during the processing of an NCR/DR, information is determined to be incorrect or inadequate, the NCR/DR shall be returned to the originator. For those DR/NCRs not being processed, the reviewer shall anotate with a statement, 'NCR/DR written in error' and a brief statement of the reviewer's reason and justification for rejecting the NCR/DR. Return DR/NCR to originator and maintain a copy in a suspense file."

This will prevent significant deficiencies from possibly being suppressed during review and any subsequent rejection of potential NCR/DRs.

The RRI (Acting) subsequently determined, that based on the information provided, the weld was acceptable and the alignment does not exceed code limitations.

No items of noncompliance or deviations were identified.

20. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. Unresolved items disclosed during the inspection are discussed in the paragraphs indicated below:

Paragraph 2 - Disposition of DR 1SD1471M

Paragraph 6 - Certification of QC Inspectors

Paragraph 17 - Storage Requirements for Mechanical Core Structures

21. Exit Interview

The RRIs (Acting) met with the various licensee representatives (listed in paragraph 1) on July 3, 10, 24, and on August 7, 1980, to discuss the findings of the July inspection efforts.