November 29, 2002

Mr. Lew Myers
Chief Operating Officer
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
NRC SPECIAL INSPECTION -REACTOR VESSEL HEAD REPLACEMENT - REPORT NO. 50-346/02-07(DRS)

Dear Mr. Myers:

On October 24, 2002, the US Nuclear Regulatory Commission (NRC) completed a special inspection at your Davis-Besse Nuclear Power Station. This inspection reviewed your actions to resolve Restart Checklist Item No. 2.a, associated with the adequacy of the reactor vessel head replacement and Restart Checklist Item No. 2.b associated with the adequacy of the containment vessel restoration following head replacement. Specifically, this inspection focused on review of a sample of activities as described in the "Davis-Besse Reactor Head Resolution Plan." To evaluate the implementation of this plan, our inspection included reviews and observations in three areas under your plan: (1) non-destructive examinations performed on the replacement head welds that occurred at the Midland Michigan site; (2) the American Society of Mechanical Engineers (ASME) Code data packages for the replacement head; and (3) activities associated with the temporary containment access opening and restoration. Additionally, we reviewed the examination of the original vessel head penetration nozzles that your staff conducted in accordance with your commitments to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." This report presents the results of our review.

Based on our inspection, we confirmed that: (1) adequate records were assembled to ensure that the replacement head was designed and fabricated in conformance with ASME Code requirements and that the original ASME Code Section III N-stamp remained valid; (2) the engineering evaluation associated with construction of the temporary containment access opening considered appropriate loads and demonstrated that stress in the containment shell materials would not exceed design limits; (3) the temporary containment vessel opening was restored such that the original ASME Code construction requirements were maintained; (4) the work activities to construct and restore the temporary containment opening and closure occurred in a controlled manner and in accordance with procedure requirements; and (5) that your managers demonstrated an active oversight role for the control of the contractors on the containment building temporary construction opening. Therefore, we concluded that the

“Davis-Besse Reactor Head Resolution Plan” was effectively implemented. At the conclusion of this inspection, your staff had not completed the final acceptance pressure tests for the vessel head and containment vessel. Therefore, Restart Checklist Item No. 2.a and 2.b will remain open pending completion of this testing.

In accordance with 10 CFR Part 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

John A. Grobe, Chairman
Davis-Besse Oversight Panel

Docket No. 50-346
License No. NPF-3

Enclosure: NRC Special Inspection Report
No. 50-346/02-07(DRS)

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John A. Grobe, Chairman
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	50-346/02-07(DRS)
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 North State Route 2 Oak Harbor, OH 43449
Dates:	February 20, 2002 through October 24, 2002.
Inspectors:	M. Holmberg, Reactor Inspector, Division of Reactor Safety, Region III James Belanger, Senior Physical Security Inspector, Division of Reactor Safety, Region III Donald Jones, Reactor Inspector, Division of Reactor Safety, Region III Richard McIntyre, Senior Reactor Engineer, Quality and Maintenance Section, Division of Inspection Program Management, Office of Nuclear Reactor Regulation Doug Simpkins, Resident Inspector, Division of Reactor Projects, Region III. John Jacobson, Reactor Inspector, Division of Reactor Safety, Region III
Approved by:	David Hills, Chief Mechanical Engineering Branch Division of Reactor Safety, Region III

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SUMMARY OF FINDINGS

IR 05000346-02-07; FirstEnergy Nuclear Operating Company; on 02/20-10/24/02; Davis-Besse Nuclear Power Station. Special Inspection.

This report covers an 8-month special inspection of licensee activities associated with the reactor vessel head examination and replacement. This inspection was conducted by a resident inspector, inspectors based in the Region III Office, and technical staff from the Office of Nuclear Reactor Regulation (NRR). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be “Green” or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. Inspection Findings

No findings of significance were identified.

B. Licensee Identified Findings

No findings of significance were identified.

REPORT DETAILS

Background and Event Overview

On March 6, 2002, Davis-Besse personnel notified the NRC of degradation (corrosion) of the reactor vessel head material adjacent to a control rod drive mechanism (CRDM) nozzle. This condition was caused by coolant leakage and boric acid corrosion of the head material induced by an undetected crack in the adjacent CRDM nozzle. The degraded area covered in excess of 20 square inches where the low-alloy structural steel was corroded away, leaving the thin stainless steel cladding layer. This condition represented a loss of the reactor vessel's pressure retaining design function, since the cladding was not considered as pressure boundary material in the structural design of the reactor pressure vessel. While the cladding did provide a pressure retaining capability during reactor operations, the identified degradation represented an unacceptable reduction in the margin of safety of one of the three principal fission product barriers at the Davis-Besse Nuclear Power Station (reference NRC report 50-346/02-03(DRS)).

At a public meeting held on June 4, 2002, the licensee described the reactor pressure vessel closure head (RPVCH) replacement program for the Davis-Besse Nuclear Power Station. At this meeting, the licensee discussed several options for resolving the degraded RPVCH including replacement with a vessel head from the Midland Michigan plant. The Midland Michigan plant had previously halted construction and was never completed. In a letter to the NRC dated August 9, 2002, the licensee described a plan for the replacement of the RPVCH using the Midland RPVCH. Because the Midland RPVCH was similar in design to the Davis-Besse RPVCH and was readily available, the licensee chose this option over repairing the existing RPVCH or fabricating a new RPVCH. The licensee issued the "Davis-Besse Reactor Head Resolution Plan" (DBHRP) which described the project management, planning, and execution of tasks needed to remove the replacement RPVCH from the containment building at Midland and subsequently install the replacement RPVCH on the reactor vessel at Davis-Besse. The NRC inspectors reviewed the licensee activities associated with the DBHRP during this inspection. Additionally, this inspection included review of the nondestructive examinations (NDE) of the original RPVCH nozzles that the licensee conducted in accordance with commitments to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." Given the high public interest in this subject area at Davis-Besse, and therefore the need to clearly communicate the rationale for NRC staff conclusions regarding the licensee's RPVCH replacement activities, this report documents the inspectors' observations.

4. OTHER ACTIVITIES

4OA3 Event Follow up (IP 93812)

.1 Davis-Besse Reactor Head Resolution Plan (DBHRP)

a. Inspection Scope

On May 24, 2002, the licensee issued the DBHRP, Revision 0 and on July 10, 2002, issued Revision 1 to the DBHRP. The NRC inspectors reviewed the DBHRP to evaluate the adequacy of the planned work scope and licensee staffing.

b. Observations

The DBHRP described the project management, planning, and execution of tasks needed to remove the replacement Reactor Pressure Vessel Closure Head (RPVCH) from the containment building at Midland and subsequently install the replacement head on the reactor vessel at Davis Besse. In this plan, the licensee described the activities necessary to support the RPVCH replacement which included:

- procurement and certification;
- RPVCH modifications;
- temporary fuel removal;
- access through the Davis-Besse containment opening;
- installation of new RPVCH;
- restoration, inspection and testing of the RPVCH and containment;
- storage and disposition of original RPVCH; and
- updating the design and licensing basis.

The inspectors noted that the original (Revision 0) of the DBHRP defined a contractor project team with clearly designated responsibilities including history of related work experience. The inspectors reviewed the personnel work histories for contractors and licensee personnel assigned to this project. Based on this review, the inspectors concluded that the licensee had assembled head replacement personnel with extensive experience in the nuclear industry and with extensive experience in similar engineering projects, such as steam generator replacement. The inspectors considered that the scope of the DBHRP and qualifications of project personnel were sufficient to accomplish the head replacement project. The licensee subsequently removed the specific work experience and education history of the personnel assigned to this project from the DBHRP (Revision 1), but did not substantively change the actual project organization or staffing.

c. Findings

No findings of significance were identified.

.2 Implementation of the Davis-Besse Reactor Head Resolution Plan

a. Inspection Scope

To evaluate the implementation of the DBHRP the inspectors reviewed activities in three areas under this plan:

1) From June 12, 2002, through June 14, 2002, the inspectors reviewed NDE performed on the replacement head welds that occurred at Midland, Michigan;

2) From August 20, 2002, through August 22, 2002, the inspectors reviewed the American Society of Mechanical Engineers (ASME) Code data packages and Consumers Energy 10 CFR Part 21 Report for the replacement head; and

3) From August 13, 2002, through October 24, 2002, the inspectors observed the activities and reviewed records associated with the containment building access opening and restoration.

b. Observations

The licensee purchased the Midland RPVCH from Framatome ANP, who in turn had purchased the head from Consumers Energy, the owner of the Midland, Michigan plant. The Midland, Michigan plant had not been completed (construction permit was issued in 1972 and construction work was suspended in 1984). The Midland RPVCH had not been placed in service since original fabrication, and had been stored inside the containment building at the Midland, Michigan site. To confirm that the Midland head could be used at Davis-Besse, the licensee's vendor (Framatome ANP, Inc.) performed non-destructive examination (NDE) and reviewed head fabrication documentation. Specifically, Framatome provided the licensee with the required documentation, NDE, analyses and ASME Code reconciliation necessary to ensure the original ASME Code N-stamp documentation was valid, and that the Midland RPVCH complied with applicable NRC and industry requirements. This documentation was assembled by Framatome ANP in ASME Code Section III and XI Quality Assurance (QA) data packages. Framatome ANP, Inc. notified the NRC, by letter dated September 9, 2002, that the Midland head conformed to all ASME Code Section III, Class A requirements, that the supporting documentation was valid, and that all markings and identification symbols matched the head configuration.

b.1 Consumers Energy Part 21 Report Evaluation

On May 23, 2002, pursuant to the reporting requirements of 10 CFR Part 21.21(b), Consumers Energy notified the NRC by letter, that the status of the Midland head was indeterminate, because the Midland reactor head had been in storage since 1986 without any routine maintenance or any oversight of a formal QA program.

The inspectors reviewed preliminary safety concern PSC 3-02, prepared by the licensee's vendor (Framatome ANP) to address this issue. In this document, Framatome ANP identified the concern, the cause, and corrective actions. The completed corrective actions included an in-depth review of the original NSS-13 QA Data Package for the reactor vessel closure head, verification of Code markings, full visual and non-destructive re-examination of all the vessel head welds and completion of the ASME Code Section XI pre-service NDE.

On September 12, 2002, Framatome ANP informed the NRC by letter, that the reactor vessel head intended for use at Midland Unit 2 was manufactured in accordance with the requirements of ASME Code Section III, Class A, 1968 Edition with Summer 1968 Addenda. In this letter, Framatome ANP asserted that the this head, with a few very minor variations, was identical to the original reactor vessel head installed at Davis-Besse, including configuration, materials, pressure and temperature rating, and vessel interface parameters.

The inspectors reviewed the corrective actions discussed in PSC 3-02 and considered the completed actions adequate to address the lack of QA controls identified in the Part 21 notification for the Midland reactor head.

b.2 Radiographic Examination of the Vessel Head Welds Conducted at Midland, Michigan

The licensee repeated radiographic examinations of the Midland head dome-to-flange weld and the control rod drive nozzle-to-flange weld to replace the original radiographic examination (RT) records, which could not be located. However, three lifting lugs on the closure head dome, spaced 120 degrees apart, prevented a complete examination of the dome-to-flange weld. The licensee subsequently determined that only 95 percent of the head dome-to-flange weld had been examined. Therefore, on August 1, 2002, the licensee submitted a letter to the NRC requesting relief (RR-A26 and RR-A27) from the ASME Code requirements to perform a 100 percent examination of this weld and to have the original RT records. The inspectors reviewed the RT records completed on these welds and did not identify any other deviations from Code requirements.

b.3 ASME Section III Data Package For The Midland Vessel Head

The inspectors reviewed the ASME Code Section III portions of the RPVCH documentation package assembled as Framatome QA Data Package No. 23-5018698-00. The data package was prepared in accordance with the Framatome ANP safety-related Quality Manual No. 56-5015885-00, which was audited and approved by the licensee.

The RPVCH was originally fabricated at the Babcock & Wilcox Mount Vernon Works for Consumer Power Company and was designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition, Summer 1968 Addenda. The RPVCH design pressure was 2500 pounds-per-square-inch-gage and design temperature was 650 degrees Fahrenheit. The RPVCH ASME Code edition, design pressure and temperature were the same as the original Davis Besse vessel head. The data package included records which demonstrated that the Midland vessel head components were stress-relieved at 1100 degrees Fahrenheit for sufficient time to meet Code requirements. The CRDM nozzles and structural J-groove welds received no post weld heat treatment in order to limit distortion, as allowed by the Section III Code requirements.

The Section III Code QA data package included rubbings of the name plate N-stamp for the reactor vessel and closure head (the “N” number is the same for both). The rubbing included the design pressure and temperature, the hydrostatic test pressure, and the date of manufacture (1975). The inspectors confirmed that this package contained records required by the Code including: the design specifications, design analyses, drawings, NDE records, hydrostatic test records and certified material test reports for pressure boundary materials.

Based on review of the this package, the inspectors concluded that sufficient records existed to confirm that the RPVCH was fabricated in accordance with the ASME Code Section III and construction QA requirements.

b.4 <u>ASME Section XI Data Package for the Midland Vessel Head</u>

The inspectors reviewed the ASME Code Section XI portion of the RPVCH documentation package assembled as Framatome QA Data Package No. 23-5019258-00. The data package was prepared in accordance with the Framatome ANP safety-related Quality Manual No. 56-5015885-00, which was audited and approved by the licensee. Additionally, the inspectors reviewed the documentation in process traveler, 50-5018614-00, for modification and preparation of the Midland RPVCH, which contained supplemental NDE records. The NDE completed by the licensee exceeded the minimum required by the Code and included:

- visual examination of the entire RPVCH to identify signs of degradation or evidence of welding while the head was in storage at Midland;
- dye penetrant examination (PT) records of all 69 CRDM J-groove welds;
- ultrasonic examination (UT) Examination of all 69 CRDM nozzles;
- PT of the head cladding at 6 sample areas;
- PT & RT of all 69 CRDM flange-to-nozzle welds;
- RT of closure head-to-flange weld;
- magnetic particle examination (MT) of the head lifting lug attachments;
- UT and MT of the closure head-to-flange weld; and
- eddy current examination of all 69 CRDM nozzle internal surfaces.

Based on review of the ASME Section XI data package and supporting documentation for supplemental NDE conducted on the RPVCH, the inspectors concluded that adequate records existed to confirm that the RPVCH was designed and fabricated in conformance with ASME Code requirements and that the original ASME Code Section III N-stamp remained valid.

b.5 <u>ASME Section XI Design Reconciliation for the Replacement Vessel Head</u>

The ASME Code Section XI required reconciliation of any differences which may exist for the replacement Code component in design, fabrication and examination requirements to ensure that the replacement component is satisfactory for the specified design and operating conditions. In a letter dated August 9, 2002, the licensee informed the NRC of the ASME Code reconciliation activities to be completed for the replacement RPVCH. In this letter, the licensee included a summary table for ASME Section XI, Article IWA-4000 Repair/Replacement Activities. The material presented in this summary table, along with the supporting vendor documents (51-5019457-00 & 01, “Davis Besse RV Closure Head Replacement Reconciliation,” and 33-5019877-00, “Davis Besse Original Closure Head Replacement Design Report,” and RPVCH drawings), provided the record of licensee activities with respect to performing the reconciliation of Code requirements.

The inspectors confirmed that the “Davis Besse Original Closure Head Replacement Design Report” was prepared, reviewed and approved by qualified personnel and was certified by two registered professional engineers, who specialized in ASME Section III Code stress analysis. Further, the inspectors confirmed that the registered professional engineers performed independent design reviews. Therefore, the inspectors concluded that the Design Report and supporting documents provided an adequate basis for the ASME Section XI reconciliation of the RPVCH.

b.6 Containment Access Opening

The Davis-Besse containment lacked an access opening of sufficient size to permit removal of the old RPVCH and reinstallation of the new RPVCH. Therefore, the licensee cut a temporary access opening in the shield building and containment vessel of sufficient size to support the RPVCH replacement. The licensee performed a detailed engineering evaluation of the work activities associated with construction of the containment access opening as documented in engineering work request (EWR) 02-0146. This package included a Design Report, in which the licensee evaluated the design requirements applicable to the containment work activities. In this report, the licensee reviewed design requirements applicable to the construction of containment vessel and shield building access openings, temporary containment reinforcement, head rigging and transport, ventilation, and restoration of the construction openings. The licensee did not perform a 10 CFR 50.59 safety evaluation for activities associated with this containment access opening because it was considered a maintenance activity, that did not change the design.

In EWR 02-0146, the licensee identified supporting calculations and applicable design requirements for the construction openings in the metal containment vessel and the concrete containment shield building. The inspectors reviewed calculation 12501-C-003, “Evaluation of Containment Vessel for Construction Opening,” which confirmed adequate structural integrity for a 20 feet wide by 20 feet high square opening in the containment vessel. The inspectors noted that this finite element calculation bounded the size of the actual containment vessel construction opening, which was 13 feet high and 18 feet wide. This calculation considered appropriate loads for the de-fueled plant conditions, which included, seismic, tornado induced, dead weight and rigging loads. The calculation demonstrated that for five load combinations, which included polar crane dead loads combined with seismic induced loads, stress in the containment shell materials would not exceed Code design limits.

The inspectors observed the following activities associated with construction of the temporary access opening in containment.

b.6.1 Shield Building Access Opening

The inspectors observed the licensee contractor cutting the construction access opening in the shield building to support the vessel head replacement. The access cut in the containment shield building was 16.5 feet high and 21.5 feet wide. The opening was made by the licensee contractor using a hydro demolition technique (high pressure water jet process) to remove the concrete. This high pressure water jet process left the original rebar intact and undamaged. The inspectors noted that even the fine rebar tie wire remained intact. The licensee contractor followed the work order 02-003545-10 which implemented the contractor Work Plan and Inspection Record (WPIR) C-CRA-02, "Cut Vessel Plate." The licensee contractor changed the cutting process from a saw cut to a torch cut for removal of the rebar in the shield building wall. The licensee approved this change based on a contractor demonstration of the capability of the torch cutting process to maintain under the maximum allowed 1/4 inch rebar cut gap.

b.6.2 Containment Vessel Access Opening

The inspectors observed the licensee's contractor performing demonstration cuts for the containment vessel on a containment mockup. The contractor used a large flat vertical steel plate with the same thickness as the containment vessel for this mockup. The contractor used an oxyacetylene torch head mounted on a motorized track assembly to produced an accurate and repeatable cut line on the mockup plate. The contractor had welding personnel demonstrate proficiency by performing several practice cuts on the mockup plate. The inspectors also observed the welders completing welds that attached steel I-beams to the mockup plate. These welds were intended to simulate conditions during the installation of reinforcement I-beams added to the periphery of the access cut to stiffen the containment vessel.

The inspectors observed the licensee contractor performing torch cutting of the containment access opening using a track mounted cutting torch. The licensee's contractor followed the work order number 02-003545-013 and WIPR C-CLP-02 controlling this process to cutout the rectangular 13 feet high by 18 feet wide access plate in the 1.5 inch thick containment vessel. The inspectors did not identify any deficiencies in the containment cutting process observed.

b.7 Containment Access Opening Restoration

The licensee reinstalled the plate section removed from the containment vessel for the temporary construction opening using a manual shield metal arc welding (SMAW) process. The licensee reinstalled this 1.5 inch thick plate with full penetration butt welds, such that the original ASME Code (Section III, of the 1968 Edition, 1969 Summer Addenda) vessel construction requirements were met.

For the shield building, the licensee reinstalled original rebar removed during the construction of the access opening and poured new concrete to close the shield building. The reinstallation of rebar and concrete conformed to original design requirements except for the requirement to test samples of the production cadweld splices as discussed below.

b.7.1 Welder Qualifications for Containment Closure Welding

The inspectors observed the licensee contractor performing qualification of welders used to fabricate the containment access closure weld. The inspectors reviewed the RT film of welds performed by seven welders during the qualification process. The contractor applied qualification weld acceptance criteria in accordance with procedure 96-RT-005, "General Radiographic Procedure Per ASME Section V Article 2." The inspectors confirmed that this procedure contained welder qualification requirements and acceptance criteria which were consistent with the requirements of the ASME Code, Section IX, 2001 Edition. The licensee's contractor had made conservative decisions in applying weld acceptance criteria for qualification welds. The licensee's contractor followed Code requirements, which included retesting welders with initial unsatisfactory welds. Two of the seven welders failed to produce satisfactory welds for this manual SMAW process and were not qualified by the licensee's contractor.

b.7.2 Containment Vessel Access Opening Closure Welding and Radiographic Examination

The NRC inspectors observed welding of the root pass on the closure plate to the access opening in the containment performed by the licensee's contract welders. The inspectors confirmed that the welding electrodes E7018, 1/8 inch diameter (designated as PCI 3229) used during this activity were of the correct material with appropriate Code records (e.g. Certified Material Test Reports). The licensee's contractor properly stored weld electrodes in a holding oven at 270 degrees Fahrenheit. The required base-metal preheat temperature was measured and confirmed by licensee Quality Control (QC) inspectors to be within acceptable range (294 degrees Fahrenheit). The licensee's QC inspectors also measured the heat input parameters (amperage, voltage and travel speed) used during the welding activities using calibrated meters. Based on these measured values (for welder M-991), the weld heat inputs used were well below the maximum of heat input of 130,909 Joules per inch required by the procedure (1 MN-GTAW/SMAW-1). The NRC inspectors concluded that the overall weld quality appeared good and that the activities were being appropriately monitored by licensee QC inspectors. Additionally, the NRC inspectors confirmed that welders observed were qualified in accordance with Code requirements.

The licensee's contractor performed SMAW of the containment closure plate, in accordance with weld procedure 1 MN-GTAW/SMAW-1. The inspectors confirmed that this welding procedure met qualification requirements from the ASME Code Section IX and impact testing requirements as specified in the original construction Code (ASME Code Section III, 1968 Edition, Summer 1969 Addenda). The containment access opening closure weld, was fabricated such that the original Code requirements (e.g. basemetal thickness and minimum preheat requirements) were met allowing the licensee to exempt a post weld heat treatment on the containment vessel.

The inspectors reviewed RT records of the containment closure welds to confirm that the these records met ASME Code acceptance criteria. The inspectors identified minor porosity and slag which had not been recorded on the reader sheets for these RT records. The inspectors also identified a base metal indication adjacent to weld CS-01D, view 48-60, which had not been recorded or evaluated. The licensee staff subsequently visually verified the indication as an acceptable surface indication and noted this on the reader sheet. An indication on weld CS-01C, view 156-168 was documented as “surface contour,” however visual verification was again not recorded on the reader sheet. The licensee subsequently visually verified this surface indication and annotated this on the reader sheet. The inspectors considered these examples to constitute minor documentation discrepancies. The key quality requirements for these RT records such as selection/placement of the penetrameter and the readily visible 2T penetrameter hole were in accordance with Code requirements. Overall, the inspectors concluded that the quality of the RT records was good, weld interpretation was generally conservative, and that indications identified were well within Code acceptance limits.

b.7.3 Welder Qualification for Shield Building Rebar

The inspectors observed the licensee contractor performing qualification of two welders for shield building reinforcing bar (rebar) welds using number 8 and number 11 rebar. The contractor used procedure P1-Rebar (0.64 CE) with E9018 filler material and a SMAW process to fabricate the qualification welds. The qualification testing included pull tests and acid etch testing as specified by the American National Standards Institute/American Welding Society D1.4-98, “Structural Welding Code-Reinforcing Steel.” The inspectors confirmed that acceptance criteria for this qualification testing met Code requirements. The inspectors noted that the licensee was using paragraph 6.1.2.2, of D1.4-98 which allowed using this Code instead of the previous Code American Welding Society D.12.1 referenced in Section 3.8.2.7 of the Updated Final Safety Analysis Report (UFSAR). The licensee had appropriately reviewed this change to the UFSAR referenced Code as documented in the 10 CFR 50.54f screening dated August 24, 2002.

The inspectors observed the licensee contractor performing qualification tests of four cadwelders using number 8 and number 10 rebar. The contractor used procedure CP-C-2 “Cadweld Rebar Splices” with a ferrous filler materials designated PBF 105 for number 10 rebar and PBF 70 for the number 8 rebar. The inspectors identified that the licensee did not have the vendor documentation on the job site which confirmed which material should be used with a given size of rebar. Specifically, the licensee was using longer sleeves with a different filler number referenced than the standard configurations identified on the vendor table. The licensee’s contractor reportedly had discussed this configuration with the cadweld vendor, but did not have documentation from the cadweld vendor accepting the specific configuration that was being used. This issued prompted the licensee to issue a stop work order for the contractor that remained in affect until improvements in management and QA oversight were implemented. The licensee subsequently contacted the cadweld vendor and obtained documentation to confirm that the correct cadweld splice and filler material configuration was being used. The licensee documented this issue in nonconformance report number 009, condition report (CR) 02-05486 and CR 02-05548.

Tensile testing of a sample of the production cadweld splices was required during the original construction of the shield building (reference Appendix 3B of the UFSAR). The licensee did not conduct testing of the production cadweld splices for the rebar reinstalled during restoration of the temporary containment shield building construction opening. Instead, the licensee chose to conduct tensile tests of cadwelds performed on removable "sister" splices, which are made using the same method and at the same location. The licensee performed an evaluation under requirements of 10 CFR 50.54 which allow changing the plant's QA program requirements. In this evaluation, the licensee concluded the change to adopt the 1995 Section III Code, Paragraph CC-4333.5.2, requirement to test "sister" splices, instead of production cadweld splices, did not constitute a reduction in commitments. This conclusion was based on NRC review and approval for this alternative in support of the D.C. Cook steam generator replacement project (reference NRC safety evaluation dated November 7, 2000). The inspectors discussed this application of 10 CFR 50.54 requirements with NRR staff and no deficiencies were identified.

b.7.4 Containment Shield Building Concrete Restoration

The inspectors observed the delivery and placement of concrete used to restore the access opening in the shield building. The licensee contractor performed this evolution in accordance with WP&IR C-SWR-01, "Shield Building Restoration." The inspectors noted that this activity was observed by licensee and contractor QA personnel. The licensee QA personnel questioned the assumptions made by the contractor regarding the reduction in air content as the concrete was pumped from the truck to the point of placement. This prompted the licensee contractor to take additional samples to confirm these assumptions. The inspectors considered that this action demonstrated an active oversight role by the licensee.

The concrete used for shield building restoration was required to meet acceptance criteria of Specification 12501-C-321, "Technical Specification for Purchase of Safety Related Ready Mix Concrete," for slump and air content at the point of delivery. After 1/3 of the first concrete truckload was placed inside the forms for the access opening, the licensee contractor made the measurements for slump and air content used to accept the concrete. The contractor measured the air content at 2.8 percent, which was below the required range of 3 to 6 percent. The licensee contractor subsequently resampled the concrete from the same wheelbarrel used for the first sample, and got an acceptable reading of 3.4 percent. The licensee issued nonconformance report number 017 to record the initial out of specification reading. The inspector noted that in accordance with ASTM C 94/C 94M-00, "Standard Specification for Ready-Mix Concrete," paragraph 16.6, if the second sample had been outside specified limits, "the concrete shall be considered failed." Because the second sample passed, the licensee considered the concrete acceptable. However, the licensee conservatively chose to not install the remaining 2/3 of the first truckload of concrete. The inspectors discussed this issue with cognizant NRR staff and no technical concerns were identified. Additionally, the inspectors reviewed the licensee's vendor report 150-20129-34, "Report of Tests on Cylinder Compressive Strengths," which documented the shear strength of the concrete cylinder samples from concrete used in restoration of the shield building. In this report, the licensee's vendor documented the that these concrete cylinder samples had compressive shear strength in excess of 5000 psi after only 7 days. This value exceeded the minimum 4000 psi minimum strength required at 28 days as discussed in Section 3.8.2.7, “Materials,” of the UFSAR. Therefore, the inspectors concluded that the strength of the concrete used in the containment shield building restoration exceeded the minimum design requirements.

After removal of the concrete forms, the licensee identified several voids exposing rebar at six areas on the inside face of the shield building wall and two areas on the outside face. These areas were typically near the top of the construction access opening and the deepest void measured 8 inches in depth (reference CR 02-07472 and 02-07080). On October 3, 2002, NRR and Region III staff held a tele-conference with licensee personnel to discuss the cause and corrective actions for this condition. The licensee stated that the voids observed were caused by air trapped at the top of the construction opening that prevented a complete fill. In addition, the licensee identified areas where the concrete surface had a rough “honeycomb” texture. The licensee stated that the honeycomb areas were caused by inadequate vibration of concrete in areas between the forms and rebar mats. The licensee’s planned corrective actions included chipping the honeycomb/voids back to sound concrete and filling the cavity with a concrete grout. The inspectors confirmed that the licensee corrective actions proposed for the voids was consistent with the governing procedure CP-C-1, “Concrete Operations.” The licensee staff concluded that no internal voids could exist because of the adequate consolidation (e.g., no trapped air) of the concrete. This conclusion was based on the relatively large area between the inner and outer rebar mats which provided adequate access for the concrete vibration tools used to consolidate the concrete.

b.8 Oversight of Containment Construction Opening and Restoration

The licensee managers and QA personnel performed independent observation of contractor activities associated with construction and restoration of the containment access opening. The QA observations included contractor activities which occurred on backshifts and weekends. Based on these observations, the licensee identified lapses in contractor oversight of work activities related to the construction of the temporary containment opening. For example, the licensee QA personnel identified that the contractor had not met acceptance criteria during a trial test run for concrete delivery to the site and that the contractor was not providing quality control personnel to monitor backshift evolutions (CR 02-05108). Additionally, the NRC inspectors identified a lack of documentation associated with a nonstandard configuration used in the qualification of cad-welders. In response to these issues, the licensee initiated a stop work order (CR 02-05548) until the contractor placed additional quality control oversight on work activities. These actions indicated that the licensee was actively engaged in oversight of contractor activities associated with construction and restoration of the containment access opening.

c. Findings

No findings of significance were identified.

.3 Reactor Head Replacement Project Security Measures

a. Inspection Scope

During the baseline inspection conducted from July 29, 2002, through August 2, 2002, the inspectors reviewed security plans onsite with the licensee Security Manager and observed the areas where the additional physical protection measures would be established.

On July 8, 2002, the licensee provided the inspectors a detailed description of the security measures planned to address the Reactor Head Replacement Project via secure telephone. The inspectors evaluated the adequacy of these measures.

b. Observations

The inspectors concluded that the additional physical protection measures appeared to be consistent with the licensee’s plans and provided appropriate interim security measures. Further, the licensee's description and plans for the security measures to address the Reactor Head Replacement Project appeared to be appropriate and well designed.

c. Findings

No findings of significance were identified.

.4 Conclusions on Reactor Vessel Head Replacement Activities

The licensee records were adequate to confirm that the RPVCH was designed and fabricated in conformance with ASME Code requirements and that the original ASME Code Section III N-stamp remained valid. Further, the licensee’s vendor corrective actions were adequate to resolve the lack of QA controls identified in the 10 CFR Part 21 notification for the RPVCH.

The licensee performed a detailed engineering evaluation of the work activities associated with construction of the temporary containment access opening which supported the head replacement. This evaluation included a calculation for the construction opening which considered appropriate loads and demonstrated that stress in the containment shell materials would not exceed design limits.

The licensee restored the temporary containment vessel construction, such that, the original ASME Code construction requirements were maintained. The inspectors confirmed that the licensee staff adhered to Code requirements during welder qualifications and containment closure welding.

The licensee work activities to construct and restore the temporary containment opening and closure occurred in a controlled manner and in accordance with procedure requirements.

The licensee managers demonstrated an active oversight role for the control of the contractors on the containment building temporary construction opening. Specifically, the QA personnel performed independent observation of contractor activities associated with the temporary containment access opening and initiated appropriate actions to improve contractor QA for lapses in the quality of work activities that were identified.

At the conclusion of this inspection, the licensee had not yet completed the final acceptance pressure tests for the vessel head and containment vessel. Therefore, NRC Restart Checklist Item No. 2a, associated with the adequacy of the reactor vessel head replacement and NRC Restart Checklist Item No. 2b associated with the adequacy of the containment vessel restoration following head replacement will remain open pending completion of this testing.

4OA5 Other

.1 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (Temporary Instruction(TI)- 2515/145)

a. Inspection Scope

On August 3, 2001, the NRC issued Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” in response to circumferential cracking identified in CRDM penetration nozzles at Oconee Nuclear Station Units 2 and 3, along with axial cracking in the J-groove welds of additional CRDM nozzles at these facilities and at Oconee Nuclear Station Unit 1 and at Arkansas Nuclear One Unit 1. This phenomenon raised concerns regarding the potential safety implications of the active degradation mechanism (PWSCC) and compliance with applicable regulatory requirements. Therefore, the NRC issued TI-145, to implement an NRC review of licensees’ activities in response to NRC Bulletin 2001-01. The Davis-Besse Nuclear Power Station was in the sub-population of plants (Bin 2) that have high susceptibility to vessel head penetration cracking (e.g., susceptibility ranking of less than 5 effective full power years from the Oconee Unit 3 condition).

From February 20, 2002, through March 6, 2002, the inspectors performed a review of the licensee’s activities in response to commitments made to NRC Bulletin 2001-01. To assess the licensee’s efforts in conducting an “effective” examination of the reactor vessel head penetration nozzles, the inspectors review included:

- observation of the licensee’s UT and visual examination of the reactor vessel head penetrations,
- interviews with the licensee’s contract NDE personnel,
- review of NDE procedures, and
- review of the head inspection NDE reports.

Additionally, the inspectors observed the repair activities implemented on the cracked vessel head penetration nozzles.

b. Observations

Summary

The licensee identified five penetration nozzle locations (1, 2, 3, 5 and 47) with axial crack indications. The licensee also determined that penetrations 1, 2 and 3 contained through-wall cracks based on UT. In addition, for penetration nozzle number 2, the licensee identified a circumferentially oriented indication just above the J-weld which extended for about 35 degrees. The licensee initiated repairs on the five cracked penetrations discussed above, and during the machining process on nozzle number 3, the licensee identified movement of the nozzle. The licensee subsequently cleaned boric acid deposits from the head during investigation of this phenomena and discovered a large cavity in the vessel head. An NRC augmented inspection team performed an inspection of this issue and the NRC team's conclusions were documented in NRC IR 50-346/02-03.

Evaluation of Inspection Requirements

In accordance with requirements of TI-145, the inspectors evaluated and answered the following questions:

a. Was the examination:

1. Performed by qualified and knowledgeable personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

Top of Vessel Head Visual Examinations
Yes. The licensee conducted remote visual examination of the head with knowledgeable personnel certified to Level II or III as VT-1 and VT-2 examiners in accordance with programs meeting the American Society for Nondestructive Testing (ASNT) Recommended Practice SNT-TC-1A and CP 189.

UT of Penetration Nozzles
Yes. The licensee conducted UT with personnel certified to Level II and Level III in accordance with programs meeting ASNT Recommended Practice SNT-TC-1A and CP 189. A portion of the licensee's UT personnel also had Electric Power Research Institute Performance Demonstration Initiative qualifications which met ASME Code Section XI, Appendix VIII requirements. Further, the lead UT analyst had experience analyzing CRDM penetration UT data at the Oconee Units.

2. Performed in accordance with approved and adequate procedures?

Top of Vessel Head Visual Examinations
Yes. The licensee conducted visual examinations in accordance with procedure 54-ISI-367-03, "Procedure for Visual Examination for Leakage of Reactor Vessel Head Penetration." The licensee's visual inspection scope included all vessel head penetrations and the visual examination method met visual quality

standards established for remote VT-1 examinations as defined in Section XI of the ASME Code.

<u>Ultrasonic Penetration Examinations</u>
Yes. The licensee's contractor conducted UT in accordance with procedure 54-ISI-100-08, "Remote Ultrasonic Examination of Reactor Head Penetrations." This procedure included instructions for UT equipment setup, calibration and sizing of indications. The licencee's contractor performed an on-site demonstration of the effectiveness of this procedure at detecting PWSCC using RPVCH penetration nozzles (removed from an Oconee Unit) that contained PWSCC.

3. Adequately able to identify, disposition, and resolve deficiencies?

<u>Top of Vessel Head Visual Examinations</u>
No. Due to the presence of boric acid and corrosion deposits, the licensee was unable to inspect 12 CRDM nozzle locations. The remaining penetrations were partially obscured such that none of the penetrations could be positively excluded as a potential source of RCS leakage. The licensee subsequently removed the boric acid deposits and identified a large cavity around penetration nozzle number 3.

<u>Ultrasonic Penetration Examinations</u>
Yes. The licensee's contractor performed UT system calibrations at 12 hour intervals on calibration standards which contained outside diameter notches. These UT examinations were conducted from the inside of the penetration and data was recorded from at least 1 inch above the nozzle J-weld groove weld to the end of the penetration tube. The licensee examined each vessel head nozzle penetration tube with a blade type UT probe. This probe head contained UT transducers setup for time-of-flight-diffraction oriented such that it provided maximum sensitivity for circumferentially oriented cracks near the outside diameter of the tube. Based on this examination, the licensee's contractor identified six penetrations with flaw/crack type indications. The licensee subsequently used a rotating head UT probe installed from above the head at each of the six penetrations with indications. The rotating UT probe contained several transducers set up for time-of-flight-diffraction which were designed to maximize response of cracks oriented in both the circumferential and axial direction. Additionally, this probe contained a 0 degree and 60 degree shear wave transducers. Based on the UT examination using the rotating probe, the licensee confirmed that cracks existed in five of the six penetrations identified by the blade UT probe. The licensee subsequently initiated repairs on these five cracked penetration nozzles.

4. Capable of identifying the primary water stress corrosion cracking phenomenon described in the bulletin?

<u>Top of Vessel Head Visual Examinations</u>
No. The vessel head nozzle penetrations were obscured by boric acid and corrosion deposits such that the licensee could not exclude any nozzle from having potential RCS leakage.

Ultrasonic Penetration Examinations

Yes - for the vessel head penetration nozzle tubes. The licensee used an UT technique demonstrated on nozzle penetration tubes removed from the Oconee Nuclear Power Station, to be effective for identifying PWSCC in the penetration tube materials. Further, the licensee used a calibration standard of similar material and dimensions as the head penetration tubes. This standard contained both axial and circumferential oriented notches located at the outside surface. Therefore, the inspectors concluded that the UT method used would be effective at detecting PWSCC in the penetration nozzle tubes.

No - for the J-welds. The UT technique used by the licensee was not designed to detect PWSCC within the J-weld region attaching the nozzle to the RPVCH. The primary inspection technique (blade probe) relied on a pitch-catch type UT method, in which the crack interferes with the sound path reflecting off the back-wall of the nozzle tube. However, at the J-weld location, PWSCC could exist beyond the back-wall of the nozzle which would not be within the sound path demonstrated as effective for detection of PWSCC. Therefore, the inspectors concluded that PWSCC could not reliably be detected if it was entirely contained in the J-weld region attaching the penetration nozzle tube to the vessel head.

b. What was the condition of the reactor vessel head (debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

Top of Vessel Head Visual Examinations

The reactor head was covered with reflective metal insulation panels installed on a support structure over the top of the reactor head. The licensee conducted the remote camera visual inspection under the insulation support structure using a camera mounted to a pole and other cameras mounted to a remote crawler. The as-found head condition was not sufficiently clear of boric acid deposits to determine if these deposits may have been the result of RCS leakage through cracked RPVCH nozzles.

Ultrasonic Examinations

The surface of the inner bore of the penetration nozzle tubes was sufficiently smooth, such that the UT was not affected and the licensee was able to achieve full coverage of each penetration nozzle.

c. Could small boron deposits, as described in the bulletin, be identified and characterized?

Top of Vessel Head Visual Examinations

No. The inspectors observed deposits of boric acid and corrosion products at each nozzle that precluded a meaningful determination of which nozzles could be sources of RCS leakage. Therefore, the licensee relied on the effectiveness of the UT to detect nozzle cracking and associated RCS leakage.

d. What materiel deficiencies (associated with the concerns identified in the bulletin) were identified that required repair?

Of the six penetrations that the licensee identified as having UT indications, five were selected for repair. For penetration nozzle number 58, the UT blade probe had detected a small axial indication. However, the licensee used the top-down rotating UT probe to confirm that cracking was not present in this nozzle and thus, did not require repair. For the remaining five penetration locations (1, 2, 3, 5 and 47) the licensee identified axial crack indications. The more significant axial crack indications typically traversed the full width of the J-weld. In addition, for penetration nozzle number 2, the licensee identified a circumferentially oriented indication just above the J-weld which extended for about 35 degrees. The licensee concluded that the axial crack indications in penetrations 1, 2, and 3 were through-wall based on analysis of the UT data and reported this condition to the NRC on February 27, 2002 (in notification number 38732). The specific number and orientation of cracks in each nozzle was documented in NRC inspection report (IR) 50-346/02-03.

The licensee initiated repairs on the five cracked penetrations discussed above. This repair process included roll expanding the penetration nozzle, grinding out the affected portion of the penetration nozzle to a location above the J-weld and addition of a temper bead weld metal buildup beginning at machined nozzle end-prep. The next step included finish machining on the inside bore, followed by UT and PT examinations of the weld. However, on March 5, 2002, during the machining process on nozzle number 3, the licensee identified movement of the nozzle. The licensee subsequently cleaned boric acid deposits from the head during investigation of this phenomena and discovered a large cavity in the vessel head. An NRC augmented inspection team performed an inspection of this issue and the NRC team's conclusions were documented in NRC IR 50-346/02-03. The NRC findings associated with this issue were documented in NRC IR 50-346/02-08. The licensee subsequently decided to replace the vessel head as discussed in Section 4OA3 of this report.

e. What, if any, significant items that could impede effective examinations and/or As-Low-As-Reasonably-Achievable issues were encountered?

The inspectors did not identify any significant impediments to the UT conducted from below the head as discussed above. The licensee's initial visual examination of the head penetrations was not effective as discussed above because of boric acid and corrosion products on the surface of the head. The licensee stated that the actual dose received for this job was about 3.5 Roentgen Equivalent Man (REM). This dose was below the licensee's projected dose for the ultrasonic examinations on the head of 5.5 REM.

c. Findings

No findings of significance were identified. However, the NRC performed additional followup inspection in this area and the results are discussed in NRC IR 50-346/02-03 and NRC IR 50-346/02-08.

.2 <u>Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (TI 2515/150)</u>

The objective of TI-150 was to review licensees' activities in response to NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs." This TI implements the NRC inspections needed to confirm that the licensee meets vessel head examination commitments associated with Bulletin 2002-02, including procedures, equipment, and personnel demonstrated to be effective in the detection and sizing of PWSCC in vessel head penetration nozzles. The Davis-Besse replacement RPVCH had never been operated, and thus, had not been exposed to the hot plant operating environmental conditions necessary to initiate PWSCC. Therefore, PWSCC does not currently exist in the replacement vessel head penetration nozzles and the NRC will complete TI-150 during the next scheduled Davis-Besse RFO.

4OA6 <u>Meetings</u>

.1 <u>Exit Meeting</u>

The inspectors presented the inspection results to Mr. L. Meyers and other members of licensee management at the conclusion of the inspection on October 24, 2002. The inspectors asked the licensee whether any materials discussed as potential report material should be considered proprietary. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee
L. Myers, Vice President - Nuclear
L. Pearce, Vice President - Oversight
R. Fast, Plant Manager
D. Baker, Project Manager
J. Reddington, Supervisor Quality Assurance
S. Loehlein, Manager Quality Assurance
A. Alford, Regulatory Affairs
S. Saunders, Senior Engineer
T. Swim, Engineer
J. Cunnings, Supervisor Mechanical Engineering
T. Chambers, Containment Health Manager
R. Mende, Containment Health Engineer

Vendor - Bechtel
S. Fox, Senior Project Manager

Vendor - Framatome ANP
E. Mayhew, Vice President Quality, US Region
V. Montalbano, Manager, Nuclear Services Quality
T. Werner, Lead Quality Specialist
M. Gerlich, QA Engineer
M. Morgan, Manager, Quality Assurance Audits and Programs
S. Dasgupta, QA Consultant
H. Behnke, Technical Consultant, Component Engineering
F. Snow, Project Engineer
H. Harrison III, Engineer

Nuclear Regulatory Commission
C. Thomas, Senior Resident Inspector
D. Simpkins, Resident Inspector
A. Mendiola, Project Manager, NRR
J. Ma, Division of Engineering, Civil Engineering and Mechanics Section, NRR
A. Ashar, Division of Engineering, Component and Containment Reliability Section, NRR

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

None

Discussed

None

LIST OF ACRONYMS USED

ASME	American Society of Mechanical Engineers
CAC	Containment Air Cooler
CFR	Code of Federal Regulations
CR	Condition Report
CRDM	Control Rod Drive Mechanism
DBHRP	Davis-Besse Head Resolution Plan
DHR	Decay Heat Removal
EWR	Engineering Work Request
MT	Magnetic Particle Examination
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PT	Dye Penetrant Examination
PWSCC	Primary Water Stress Corrosion Cracking
RE	Radiation Element
QA	Quality Assurance
QC	Quality Control
REM	Roentgen Equivalent Man
RCS	Reactor Coolant System
RPVCH	Reactor Pressure Vessel Closure Head
RT	Radiographic Examination
SDP	Significance Determination Process
SMAW	Shield Metal Arc Welding
TI	Temporary Instruction
UT	Ultrasonic Examination
UFSAR	Updated Final Safety Analysis Report
WPIR	Work Plan and Inspection Record

LIST OF DOCUMENTS REVIEWED

ASME Code Data Packages

F-ANP QA Data Package No. 23-5018698-00	Midland Replacement Reactor Pressure Vessel Closure Head ASME B&PV Code Section III Data Package	Revision 0
F-ANP QA Data Package No. 23-5019258-00	Midland Replacement Reactor Pressure Vessel Closure Head ASME B&PV Code Section XI Data Package	Revision 0

Calculation

12501-C-003	Evaluation of Containment Vessel for Construction Opening	Revision 1

Condition Reports

6015661	Supplemental examinations needed to aid in the evaluation of radiography film.
02-00891	Reactor Vessel Head
02-00932	Reactor Vessel Head
02-01053	Number 3 nozzle machining tool moved approximately 15 degrees
02-05108	Bechtel Quality Assurance Oversight Concerns
02-05548	Issues have been raised regarding the effectiveness of Bechtel Quality Assurance program that reveal a negative trend
02-05486	The testing and qualification of cadwelders for the restoration of containment was started without written direction of the vendor.
02-07472	Containment Shield Building-Annulus Side
02-07080	Containment Shield Building

Drawings

12501-C-102	Reactor Pressure Vessel Head Replacement Project Containment Shield Building Construction Opening Details	Revision 2
12501-C-103	Reactor Pressure Vessel Head Replacement Project Containment Shield Building Barrier and Lug Details	Revision 4
12501-C-201	Reactor Pressure Vessel Head Replacement Project Temporary Opening in Containment Vessel Wall	Revision 4

F-ANP 5018608B-01	Midland 2 RVH CRDM nozzle modification	Revision 1
F-ANP 6015305B-1	CRDM nozzle flange modification go gauge	Revision 1
F-ANP 6015470B-1	CRDM nozzle flange modification no go gauge	Revision 1
F-ANP 5018780B-2	Modified CRDM flange nut ring installation	Revision 2
F-ANP 5018900B-1	Midland head x-axis keyway block modification	Revision 1
F-ANP 5018532E-0	Midland -2 service structure support skirt openings	Revision 0

Engineering Work Request

02-0146	Provide Opening in the Containment Structure to Remove/Replace the Reactor Vessel Head	Supplement 2

Field Change Requests

FCR-C-003	EWR 02-0146	August 8, 2002
FCR-C-008	EWR 02-0146	September 6, 2002
FCR-C-016	EWR 02-0146	September 22, 2002
FCR-C-017	EWR 02-0146	September 23, 2002
FCR-C-019	EWR 02-0146	September 07, 2002
FCR-C-022	EWR 02-0146	September 14, 2002
FCR-C-027	EWR 02-0146	September 23, 2002
FCR-C-028	EWR 02-0146	September 24, 2002

Inspection Plans

	Davis-Besse Reactor Head Resolution Plan	Revision 0
	Davis-Besse Reactor Head Resolution Plan	Revision 1

Nonconformance Reports

NCR-009	CP-11	September 4, 2002
NCR-017	Specification 12501-C-321	September 24, 2002

Other Documents

21896-002	Mockup of Containment Cutting for Welding	Revision 0
Work Order 02-003545-010	A Temporary Access Opening in the Containment Shield Building is Required to Support Replacement of the RPV Head	August 10, 2002
WIPR C-CRA-02	Removal of Concrete in the Temporary Construction Opening Through the Containment Shield Wall	Revision 0
Work Order 02-003545-013	A Temporary Access Opening in the Containment Shield Building is Required to Support Replacement of the RPV Head	August 21, 2002
WIPR C-CLP-02	Cut Vessel Plate	Revision 0
50-5018614-00	Process Traveler Modification and Preparation of Midland-2 RVCH	June 4, 2002
54-PT-6-07	Visible Solvent Removable Liquid Penetrant Examination Procedure	August 3, 2000
54-1027734-05	Radiographic Testing	June 3, 2002
	Midland Reactor Pressure Vessel Control Rod Drive Mechanism Dissimilar Weld Radiography Supplemental Examination Plan	June 12, 2002
02-0146-00	Provide access opening in the containment structure to remove/replace the reactor vessel head	August 2, 2002
Letter from J. Mallay (Framatome ANP INC.) to NRC	Closure of Interim Report Concerning a Potential Safety Concern on the Condition of the Midland Reactor Vessel Head Proposed for Use at an Operating Plant	September 9, 2002
Nonconformance report 009	Cadweld Qualification Testing	September 4, 2002
WP&IR C-SWR-01	Shield Building Restoration	Revision 0
Bechtel Specification 12501-C-321	Technical Specification for Purchase of Safety Related Ready-Mix Concrete	Revision 2

Other Documents

	Davis Besse Nuclear Power Station Reactor Pressure Vessel Head Replacement Project Concrete Placement and Test Plan	Revision 0
Certified Material Test Report	Atom Arc 7018, 1/8," 9950 lbs, lot number 4G113A07	August 21, 2001
PSI report 150-20129-34	Report of Tests on Cylinder Compressive Strengths	October 1, 2002
	David Besse 13 RFO CRDM Nozzle Examination Report	March 11, 2002
Letter Serial Number 1-128	FirstEnergy letter to Mr. James E. Dyer, Administrator - Replacement of the Reactor Pressure Vessel Head at the Davis Besse Nuclear Power Station	August 9, 2002
F-ANP Document 03-5018636-00	CRDM housing flange modification drill fixture operating instruction, Midland 2	Revision 0
F-ANP Technical Document 33-5019877-00	ASME Stress Report, Davis Besse Original Closure Head Replacement Design Report, Davis Besse Unit 1	August 15, 2002
F-ANP Document 51-5018522-10	Midland Closure Head Dedication Plan	July 17, 2002,
F-ANP Document 51-5019457-00	Davis Besse RV Closure Head Replacement Reconciliation	August 16, 2002
F-ANP Document 51-5019457-01	Davis Besse RV Closure Head Replacement Reconciliation	August 22, 2002
02-046u	UFSAR Change Notice for EWR 02-0146	August 24, 2002
Bechtel Specification No. 12501-C-310	Technical Specification for Purchase of Safety Related Ready-Mix Concrete	Revision 2
Bechtel Specification No. 12501-C-321	Technical Specification for Installation of Cadweld Rebar Splices	Revision 2
Bechtel Specification No. 12501-C-322	Technical Specification for Purchase of Safety Related Ready-Mix Concrete	Revision 2

Other Documents

Bechtel Specification No. 12501-C-101	Technical Specification for Materials Testing Services	Revision 3
	Quality Assurance Program Manual	Revision 3
Williams Concrete Inc. Letter	Mix Design Submittal Information	July 29, 2002
Bechtel Document	Concrete Placement and Test Plan	September 24, 2002
Bechtel WP&IR	Shield Building Restoration	September 14, 2002
12501-SC-025-PQR-721A-02	Procedure Qualification Record	Revision 1
12501-SC-025-PQR-627A-02	Procedure Qualification Record	Revision 3
Specification 7749-C-37	Containment Vessel Technical Specification	Revision 19
Erico Concrete Reinforcement Products Memorandum	CADWELD rebar splice configuration	September 5, 2002

Procedures

96-RT-005	General Radiographic Procedure Per ASME Section V Article 2	Revision 5
CP-C-1	Concrete Operations	Revision 0
CP-C-2	Cadweld Rebar Splices	Revision 0
CP-C-11	Testing of Cadweld Rebar Splices	Revision 0
54-ISI-367-03	Procedure for the Visual Examination for Leakage of the Reactor Vessel Head	Revision 3
54-ISI-100-06	Remote Ultrasonic Examination of Control Rod Drive Mechanism (CRDM) Nozzles	Revision 6
1 MN-GT-GTAW/SMAW-1	Welding Procedure Specification	Revision 14

96-RT-005	General Radiographic Procedure Per ASME Section V Article 2	Revision 5
CP-C-1	Concrete Operations	Revision 0
CP-C-2	Cadweld Rebar Splices	Revision 0
CP-C-11	Testing of Cadweld Rebar Splices	Revision 0
54-ISI-367-03	Procedure for the Visual Examination for Leakage of the Reactor Vessel Head	Revision 3
54-ISI-100-06	Remote Ultrasonic Examination of Control Rod Drive Mechanism (CRDM) Nozzles	Revision 6
Radiographic Records		
CS-01A	Containment Vessel Weld	September 30, 2002
CS-01B	Containment Vessel Weld	September 30, 2002
CS-01C	Containment Vessel Weld	September 30, 2002
CS-01D	Containment Vessel Weld	September 30, 2002
W-7	Reactor Vessel Closure Head to Head Flange Weld (Replacement Head)	
W-9	Control Rod Drive Mechanism NiCrFe Body-to -Stainless Steel Weld (Replacement Head)	
Surface Examination Records on Replacement Head Welds		
W-13	Dye Penetrant Record of J-groove Buttering Weld	
W-15	Magnetic Particle Record of Service Structure Segments to Closure Head Weld	
WH-17	Magnetic Particle Record of Lift Lug to Closure Head Weld	
WH-25	Dye Penetrant Record of CRDM Nozzle J-grove weld	
WH-27	Dye Penetrant Record of Arrow to Closure Head	

96-RT-005	General Radiographic Procedure Per ASME Section V Article 2	Revision 5
CP-C-1	Concrete Operations	Revision 0
CP-C-2	Cadweld Rebar Splices	Revision 0
CP-C-11	Testing of Cadweld Rebar Splices	Revision 0
54-ISI-367-03	Procedure for the Visual Examination for Leakage of the Reactor Vessel Head	Revision 3
54-ISI-100-06	Remote Ultrasonic Examination of Control Rod Drive Mechanism (CRDM) Nozzles	Revision 6

Welder Qualifications

	PCI Welder M990	Certified on August 19, 2002
	PCI Welder M991	Certified on August 19, 2002
	PCI Welder M989	Certified on August 16, 2002
	PCI Welder M987	Certified on August 13, 2002
	PCI Welder M985	Certified on August 9, 2002
	PCI Welder M983	Certified on August 9, 2002
	Bechtel Welder IW-1	Certified on August 29, 2002
	Bechtel Welder IW-3	Certified on August 29, 2002
	Bechtel Welder IW-7	Certified on August 27, 2002
	Bechtel Welder IW-8	Certified on August 27, 2002
	Bechtel Splicer 7423	Certified on September 15, 2002

96-RT-005	General Radiographic Procedure Per ASME Section V Article 2	Revision 5
CP-C-1	Concrete Operations	Revision 0
CP-C-2	Cadweld Rebar Splices	Revision 0
CP-C-11	Testing of Cadweld Rebar Splices	Revision 0
54-ISI-367-03	Procedure for the Visual Examination for Leakage of the Reactor Vessel Head	Revision 3
54-ISI-100-06	Remote Ultrasonic Examination of Control Rod Drive Mechanism (CRDM) Nozzles	Revision 6
	Bechtel Splicer 8578	Certified on September 15, 2002
	Bechtel Splicer 6792	Certified on September 15, 2002
	Bechtel Splicer 8715	Certified on September 15, 2002
	Bechtel Splicer 5251	Certified on September 15, 2002
	Bechtel Splicer 5243	Certified on September 15, 2002
	Bechtel Splicer 6906	Certified on September 15, 2002

DOCUMENTS REQUESTED

Information to provide to M. Holmberg for on-site inspection beginning on August 13, 2002.

A. For the containment vessel access cut in support of head replacement provide a copy of:

1) Detailed schedule for containment access cut and restoration including description of related activities such as welding, nondestructive testing, welder qualification and/or mockup training.

2) ASME Code repair/replacement plan identifying Construction Code and Code Cases used for the containment vessel access cut. Specifically, identify the applicable Code Section(s) and Edition applicable to the containment closure weldment and the acceptance criteria for the applicable nondestructive testing.

3) Fabrication and weld construction drawings for the containment vessel. Drawings associated with the containment access cut and restoration.

4) List identifying the weld process, procedure and applicable revision for each new weld on the containment vessel.

5) List of welders or weld operators that are to be used to perform welding on the containment vessel.

6) List of design change packages and safety evaluations associated with the containment vessel access cut.

7) List of condition reports (beginning in January of 2002) and non-conformance reports associated with containment vessel, with a brief description of the condition.

8) The containment vessel design specification and containment coating design specification.

9) Containment modification package (EWR 02-0146) including 50.59 evaluation and supporting containment vessel analysis for the temporary containment opening.

10) The welding procedures and supporting qualification documents (PQRs and test reports) used to close the temporary vessel access cut.

11) List of procedures/work orders (including description) that control the work activities and non-destructive testing.

12) Procedure(s) that identify the quality control hold point and witness checks for containment access work (installation and testing) as specified by the on-site quality control organization.

INFORMATION REQUESTED ON 12/18/2001 BY E-MAIL (To R. Cook)

A. Please provide the following information to Melvin Holmberg at the Region III NRC office located at 801 Warrenville Rd, Lisle IL 60532, no later than January 7, 2002, to support the NRC Inservice Inspection (IP 71111.08 and TI-145) scheduled at the Davis Besse plant for February 20, 2002 - March 8, 2002.

1) A detailed schedule of nondestructive examinations planned for Class 1 & 2 systems and containment, performed as part of your ASME Code ISI Program during the scheduled inspection weeks. Provide a detailed schedule of vessel head examinations which fulfill NRC commitments made in response to NRC Bulletin 2001-01. Provide a detailed schedule of steam generator (SG) tube inspection and repair activities for the upcoming outage,
2) A copy of the procedures used to perform the examinations identified in A.1. For ultrasonic examination procedures qualified in accordance with Appendix VIII, of Section XI of the ASME Code, provide documentation supporting the procedure qualification (e.g. the EPRI performance demonstration qualification summary sheets). Also, include documentation of the specific equipment to be used (e.g. ultrasonic unit, cables, and transducers including serial numbers).
3) A copy of any ASME Section XI, Code Relief Requests applicable to the examinations identified in A(1).
4) A list identifying nondestructive examination reports (ultrasonic, radiography, magnetic particle, dye penetrant, visual (VT-1, VT-2, VT-3)) which have identified relevant indications on Code Class 1 & 2 systems in the past two refueling outages (both Units).
5) List of welds in Code Class 1, and 2 systems which have been completed since the beginning of the last refueling outage (both Units and identify system, weld number and reference applicable documentation).
6) For reactor vessel weld examinations required by the ASME Code, that are scheduled during the inspection, provide a detailed description of the welds to be examined, extent of the planned examination and a copy of your responses to the NRC, associated with Generic Letter 83-15.
7) Provide a list with description of ISI and steam generator related issues entered into your corrective action system beginning with the date of the last refueling outage (both Units).
8) Copy of any part 21 reports submitted beginning with the date of the last refueling outage.
9) Copy of SG history documentation given to vendors performing eddy current (ET) testing of the SGs during the upcoming outage.
10) Copy of procedure containing screening criteria used for selecting tubes for in-situ pressure testing and the procedure to be used for in-situ pressure testing.
11) Copy of previous outage SG tube operational assessment completed following ET of the SGs.
12) Copy of the document defining the planned ET scope for the SGs and the scope expansion criteria which will be used.
13) Copy of the document describing the ET probe types, and ET acquisition equipment to be used, including which areas of the SG (e.g. top of tube sheet, U-bends) each probe will be used in. Also, provide your response letter(s) to generic letters 95-03, 95-05, 97-05, and 97-06.
14) Copy of document describing actions to be taken if a new SG tube degradation mechanism is identified.

Enclosure (1)

15) Identify the types of SG tube repair processes which will be implemented for defective SG tubes. Provide the flaw depth sizing criteria to be applied for ET indications identified in the SG tubes.
16) If tube leakage was identified during the previous operating cycle, provide documentation identifying which SG was leaking and planned corrective actions.
17) Provide a copy of the EPRI Technique Specification Sheets which support qualification of the ET probes to be used during the upcoming SG tube inspections.
18) Provide a copy of the guidance to be followed if a loose part or foreign material is identified in the SGs.
19) Detailed scope of the planned nondestructive examinations (NDE) of the vessel head which identifies the types of NDE methods to be used on each specific part of the vessel head to fulfill NRC commitments made in response to NRC Bulletin 2001-01. Also include examination scope expansion criteria and planned expansion sample sizes if relevant indications are identified.
20) Copy of NDE procedures to be used for performing vessel head inspections that fulfill NRC commitments in response to NRC Bulletin 2001-01.
21) Identify what standards or requirements will be used to evaluate indications identified during NDE examinations of the vessel head.

B. Information to be provided on-site to the inspector at the entrance meeting:

1) For welds selected by the inspector from A.5 above, provide copies of the following documents:
    a) Document of the weld number and location (e.g. system, train, branch).
    b) Document with a detail of the weld construction.
    c) Applicable Code Edition and Addenda for weldment.
    d) Applicable Code Edition and Addenda for welding procedures.
    e) Applicable weld procedures (WPS) used to fabricate the welds.
    f) Copies of procedure qualification records (PQRs) supporting the WPS on selected welds.
    g) Copies of mechanical test reports identified in the PQRs above.
    h) Copies of the nonconformance reports for the selected welds.
    i) Radiographs of the selected welds and access to equipment to allow viewing radiographs.
    j) Copies of the preservice examination records for the selected welds.

2) For the replacement activities selected by the inspector provide a copy of the records of the repair or replacement required by the ASME Code Section XI Articles IWA -4000 or IWA 7000.
3) Provide a list of NDE personnel performing inspections of the vessel head and the qualification records for these personnel.
4) Copies of commitments made to the NRC for performing vessel head examinations.
5) Copy of the most recent quality assurance department audit, which included the ISI program and activities. Copies of documents resolving findings in this audit.
6) Updated schedules for item A.1.