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ROBERT C. MECREDY
Vice President
Nuclear Operations

September 4, 2001

United States Nuclear Regulatory Commission
Document Control Desk
Attention: Robert L. Clark
Project Directorate I-1
Washington DC 20555

Subject: Response to NRC Bulletin 2001-01, Subject: CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Clark:

On August 3, 2001, the Nuclear Regulatory Commission issued the referenced bulletin requesting that all addressees provide to the NRC a written response within 30 days in accordance with the provisions of 10 CFR 50.54(f). Information requested relates to the structural integrity of the reactor pressure vessel penetration (VHP) nozzles for their respective facilities, the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and the basis for concluding that their plans for future inspections will ensure compliance with applicable regulatory requirements. The attached enclosures provide the requested information.

Very truly yours,

Robert C. Mecredy

Sworn and subscribed before me
on September 4, 2001

Sharon L. Miller
Notary Public

SHARON L. MILLER
Notary Public, State of New York
Registration No. 01MI6017755
Monroe County
Commission Expires December 21, 2002

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Enclosures 1 - 5

xc: Mr. Robert L. Clark (Mail Stop O-8-E9)
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Regulatory Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

U.S. NRC Ginna Senior Resident Inspector

## ENCLOSURE 1
## NRC BULLETIN 2001-01 REQUESTED INFORMATION

(1) *All addressees are requested to provide the following information:*

a. *the plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report;*

**RESPONSE:**

The Rochester Gas and Electric Corporation (RG&E) Ginna Station has been ranked for the potential for primary water stress corrosion cracking (PWSCC) of the reactor pressure vessel head penetration (VHP) nozzles using the time-at-temperature model and plant-specific input data reported in MRP-48 [Ref. 1].

As shown in Table 2-1 of MRP-48, this evaluation indicates that it will take 15.0 effective full power years (EFPYs) of additional operation from March 1, 2001, to reach the same time at temperature that Oconee Nuclear Station Unit 3 (ONS3) had at the time that its leaking nozzles were discovered in February 2001.

Using the criteria stated in NRC Bulletin 2001-01, Ginna Station falls into the NRC category of plants with greater than 5 EFPYs and less than 30 EFPYs until reaching the ONS3 time-at-temperature conditions.

b. *a description of the VHP nozzles in your plant(s), including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles;*

**RESPONSE:**

The Ginna Station reactor vessel head contains a total of 38 penetrations which are used for the following applications: 29 full length control rod drives, 4 part length control rod drives (inactive but left in-place), 4 instrumentation ports (of which one is a spare), and one small bore diameter head vent. The requested penetration information is provided in Table 2-3 of MRP-48 [Ref. 1].

c. *a description of the RPV head insulation type and configuration;*

**RESPONSE:**

As reported in Table 2-1 of MRP-48 [Ref 1], Ginna Station has contoured block insulation on the reactor vessel head.

The contoured block insulation on the Ginna Station reactor vessel head is called out on the Original Equipment Manufacturing (OEM) drawing (117807E) as two, one and one-half inch thick layers of B&W "KAOWOOL" block, covered with "FIBERFRAX" cloth, and covered with "FIBERFRAX" cement. The drawing also calls for the cement to be waterproofed with Silicone resin.

The original Westinghouse E-spec 676206 calls for the vessel head dome to have block type "UNIBESTOS" or equal, with voids to be filled with asbestos cement prior to application of asbestos tape, covered by a layer of "ONE COTE" cement or Eagle Picher "ONE COTE" over tape.

Note the composition of the insulation has not been field verified; therefore, the potential for equivalent substitutes such as asbestos does exist.

Enclosure 2 contains a photograph that was taken through the Lower CRDM Cooling Shroud HVAC duct connection port. The insulation can be seen in the lower area of the photograph.

d. *a description of the VHP nozzle and RPV head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations;*

**RESPONSE:**

In 1999, RG&E performed an inner diameter eddy current inspection of the Ginna Station VHPs as reported in Table 2-1 of MRP-48 [Ref. 1]. The following is additional information related to this and earlier inspections.

Following the initial discovery of the Alloy 600 concerns in the early 1990's, and prior to issuance of GL 97-01, RG&E performed a visual examination (with engineering personnel) of the penetration welds under the vessel head utilizing a remote control video camera during the 1993 refueling outage. While not a formally qualified visual inspection, this inspection did show that no gross indications existed on the surface of the accessible welds.

RG&E also performed non-qualified visual inspections on top of the insulation through the existing Lower CRDM Cooling Shroud HVAC duct connection ports. While these type of inspections have since been shown to be less than effective for the type of leakage discovered at Oconee Station, small gaps at some of the penetration/insulation interface did provide, at the time, reasonable assurance that no gross defects existed at Ginna Station.

As discussed in response to question 1c above, Enclosure 2 shows the insulation as it exists on the Ginna Station reactor vessel head. In order to gain access to remove and replace the insulation, the lower and upper CRDM Cooling Shroud would have to be disassembled and removed. Dose rates in the area of the coil stacks and shroud region would be expected to be 1 R/hr or greater.

As part of the Industry response to Generic Letter 97-01, Ginna Station voluntarily to performed internal diameter inspections of the upper head penetrations during the 1999 refueling outage. These inspections were performed utilizing Rotating Pancake Coil and Blade Probe Eddy Current (EC) technology for detection of indications on the inside surface of the head penetrations. Ultrasonic Technology (UT) was then applied to determine flaw characterization of any EC indications.

At the time the 1999 inspection was performed, the primary area of concern was the inside surface of the Alloy 600 penetration material in the region which contained the weld profile of reactor vessel head to Alloy 600 penetration J-Groove weld. The inspections were performed utilizing the ROMAN delivery system as supplied by Framatome Technologies which, in addition to using FTI and RG&E personnel, also employed several technicians who were highly experienced with the inspections previously performed in the European PWR community.

Rotating EC probes were used on the instrumentation penetrations and the head vent. Blade type EC probes were used on the remainder of the full and part length CRDM penetrations. Demonstration, qualification, and the probes used during the 1999 examinations at Ginna Station for EC and UT Interrogation are discussed in Enclosure 5.

The rotating EC probes completed 100% of the intended examinations in the 4 instrumentation nozzles and the head vent. The Blade Probe System, was designed to be inserted into the approximate 1/8 inch gap between the thermal sleeve outer wall and the Alloy 600 head penetration inner wall on the full length control rod drive penetrations. It was also used on the part length control rod drive penetrations which have no thermal sleeve. The Blade Probe achieved 93% of the intended coverage.

Results of the EC examination showed that one penetration (#13) had axially orientated, craze type indications. Subsequent UT characteristics showed "below UT detectable depth" indications. Per the established inspection criteria, this indication could have been called "no indication present", as it was below the established inspection criteria. However, as documented in our corrective action process (ACTION Report 99-0499), these indications were conservatively assumed to be at the minimum depth that the UT system was capable of detecting, or 2mm to allow futher evaluation.

Disposition of ACTION Report 99-0499 utilized a plant-specific evaluation WCAP-15143, "Structural Integrity Evaluation of Reactor Vessel Upper Head Penetrations to Support Continued Operation: R.E. Ginna". This WCAP was used to demonstrate that it would take approximately 13.7 years to reach a flaw depth of 0.75t of the penetration wall thickness. In 1999, Ginna Station had only 10 years remaining on its current Operating License which was less than the 13.7 year window. Hence, RG&E determined that even if the indication on Penetration #13 were conservatively considered a detectable indication (which was not required to be assumed), there was no concern prior to the Ginna Station current Operating License expiration in 2009.

Per NRC direction during the 1999 inspections, a copy of ACTION Report 99-0499 and its disposition was supplied to an NRC Inspector who was on site during the 1999 inspections performing other engineering type reviews (see NRC Inspection Report 50-244/99-02, Section M2.1.b).

Following the Oconee experience in 2001, RG&E requested that the vendor who performed the Ginna Station inspection in 1999, review its conclusions on the ultrasonic data for penetration #13 since the vendor had also performed inspections at Oconee. In a letter dated August 7, 2001, FTI responded to the request and concluded that a second ultrasonic data review did not change the conclusions originally reached on penetration #13.

The 1999 EC inspection focus was on axial or circumferential indications on the inside surface of the VHPs. During the development of WCAP-15143, it was postulated that the potential existed for indications other than those for which the EC inspection was being used. (e.g., visual observations during personnel entry under the head) This included the potential for an indication in the weld region. Following review, the weld crack issue was discounted due to the configuration of the weld since, even if the weld cracked, part of it would still remain attached to the penetrations and be larger then the existing reactor vessel head hole diameter and thus impede separation from the vessel. However, an additional case of a hypothetical circumferential crack on the outside surface of the VHP was identified as being a more critical case.

The acceptance criteria for this OD crack was evaluated in WCAP-15143 with a maximum allowable depth for a flaw at or above the weld of 75 percent of the penetration wall thickness regardless of flaw orientation. This 75 percent limitation was selected to be consistent with the maximum acceptable flaw depth in ASME Section XI, and to provide as additional margin against through wall penetration. The axial flaw orientation criteria was previously approved for use by the NRC at other facilities per Reference 2 (note - since circumferential indications were not anticipated due to industry experience at the time, the NRC reserved the right to review any circumferential indications discovered on a case by case basis). WCAP-15143 concluded that a significant amount of through-wall circumferential cracking would have to occur prior to creating the potential to reach plastic instability in the CRDM wall.

In addition to the inspections noted above, the lower portion of the reactor vessel head between the upper surface of the head flange and the CRDM Cooling Shroud support ring is visible to the refueling crew during each refueling outage. This region, approximately 20-24 inches in arc length, is insulated with reflective type insulation panels which are removed by the refueling crew prior to disassembly of the reactor vessel (see Enclosure 3A, by lower red ring or arrow).

Since the 20-24 inch arc length under discussion is below the outer most row of penetrations, this area could be expected to show telltale signs of boric acid leakage from outer row penetrations, and be obvious to refueling crews during reflective insulation removal. Appropriate corrective action would be taken to determine the source of boric acid leakage if this were observed.

RG&E also performs visual checks for leakage of the reactor coolant system following each refueling outage. However, the insulation on the reactor vessel upper head is not removed for these inspections.

*e.* *a description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.*

**RESPONSE:**

Enclosures 3A, B, C, and D provide a general area view of the head as it sits on the reactor vessel. Enclosure 4 is an OEM drawing for the missile shield, including relevant dimensions. Further discussion is provided below:

<u>Enclosure 3A, Reactor Vessel Head region in the refuel cavity</u> - Starting in the lower right area of the vessel (see the arrowhead by the red ring near the flange),

the reflective insulation removed each refueling outage as discussed in a previous section can be seen just below the circular HVAC duct with the FME protective cover. Moving upward, the circular HVAC duct is partially shown. In its fully installed state, this duct work encircles approximately 270 degrees the head region.

The HVAC duct work connects to the CRDM Cooling Shroud Lower Chamber (circular section) which can be seen immediately behind the circular HVAC duct work. Above and connected to the CRDM Cooling Shroud Lower Chamber is the CRDM Cooling Shroud Upper Chamber (rectangular section). Note that one instrumentation connection point (Conoseal) can be seen at the flat junction plate area above and to the left of the HVAC duct work connection to the lower CRDM Cooling shroud (light spot).

Behind the rectangular Upper CRDM Cooling Shroud are the control rod motor drive coils with the position indicating system coil stacks above. The large tube steel columns connected to the head are part of the head lift rig assembly. Note that one is located just behind the stud tensioning device lift rig at the top of the picture. The small bore pipe protruding out away from the lift rig column is part of the Reactor Vessel head vent system.

This photograph (best available) is shown with the duct work (vertical riser) and instrumentation ports partially dis-assembled.

Enclosures 3B and 3C, Missile Shield Area - Note that the majority of the cable runs and connections are located outside the periphery of the CRDM upper housing area. Cables are then bundled together for entry above the CRDM area as shown in Enclosure 3C.

Enclosure 3D shows a portion of the reactor vessel upper head seismic restraint.

(2) *If your plant has previously experienced either leakage from or cracking in VHP nozzles, addressees are requested to provide the following information:*

a. *a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;*

b. *a description of the additional or supplemental inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken in response to identified cracking to satisfy applicable regulatory requirements;*

c. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

d. *your basis for concluding that the inspections identified in 2.c will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

(1) *If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

(2) *If your future inspection plans do not include volumetric examination of all VHP nozzles, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will be satisfied.*

**RESPONSE:**

Ginna Station has not identified either leakage from, or cracking in, VHP nozzles and does not fall within this classification. Note that the craze indications in #13 were below minimum detection levels for the UT system and were not considered "cracks".

**(3)** *If the susceptibility ranking for your plant is within 5 EFPY of ONS3, addressees are requested to provide the following information:*

a. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

b. *your basis for concluding that the inspections identified in 3.a. will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

(1) *If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

(2) *If your future inspection plans include only visual inspections, discuss the corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.*

**RESPONSE:**

Ginna Station is between 5 and 30 EFPY of ONS3 and does not fall within this classification based on MRP-48 [Ref. 1].

**(4)** *If the susceptibility ranking for your plant is greater than 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following information:*

*a. your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

**RESPONSE:**

As described above, RG&E has performed several different types of inspections of the VHPs at Ginna Station. These range from checks of reactor coolant system leakage during pressurization tests following each refueling outage, informal visual inspections of both the top and underside of the reactor vessel head, to EC inspections of the internal VHPs. The latter inspection in 1999 verified no through-wall cracks for 93% of the critical surface area of VHPs at Ginna Station.

Bulletin 2001-01 describes additional types of inspections including visual inspection of the reactor vessel head near the VHPs and volumetric examination of VHP welds. However, there are issues related to standard acceptance criteria for these inspections as indicated in the bulletin. As such, RG&E plans to determine the type of inspection, if any, that will be performed during the upcoming March 2002 refueling outage. This will be provided to the NRC by December 31, 2001. The basis for the delay in providing the type of inspection and schedule is as follows:

a. Bulletin 2001-01 references the use of the EPRI MRP susceptibility model for the PWSCC of the VHPs. As described above, Ginna Station is within the moderate susceptibility category and does not meet the conditions which occurred at Oconee until 15 EFPYs after March 1, 2001. The bulletin recognizes the importance of crack growth with respect to maintaining structural integrity between examinations. This issue is under investigation by both the nuclear industry and the NRC with the expectation that more information will be available later this year. Since RG&E performed the 1999 EC inspection, the crack growth rate is an important consideration with respect to inspection frequency such that it is prudent to await these data to better support making the correct decision.

b. There are also numerous PWRs with scheduled refueling outages in the fall of 2001. These include plants with higher, lower, and similar susceptibility per the EPRI MRP model. Data obtained from these plants would assist both crack growth rate determinations and the standardization of testing acceptance criteria. RG&E believes it to be beneficial to await this information before determining the type and schedule of future inspections.

c. Dose rates at the outside of the lower CRDM cooling shroud are approximately 1 R/Hr. Dose rates inside the shroud are expected be higher. Also, the type of insulation material on top of the reactor vessel head is currently indeterminate. Since the fall 2001 plants may address some of these issues, RG&E believes it prudent to allow additional time to determine the appropriate inspection method, develop an effective approach for the removal, design and replacement of insulation (if required), and incorporate ALARA considerations.

d. As indicated in Enclosure 5, any VHP inner diameter surface breaking cracks (even if initiated from the outer diameter) would have been detected during the 1999 inspection. RG&E believes this data could potentially support continued operation until the 2003 refueling outage without further inspections. However, RG&E is continuing to evaluate the data along with data expected from the fall 2001 outages.

e. RG&E is currently pursuing reactor vessel head replacement during the Fall 2003 refueling outage. Based on current information, RG&E believes this is the best strategy to ensure long-term management of this issue as well as maintaining ALARA considerations. RG&E experience with volumetric inspections of these penetrations during the 1999 refueling outage demonstrated that they can be dose intensive and expensive. Data from fall 2001 outages and industry and NRC initiatives will help finalize this course of action.

*b. your basis for concluding that the inspections identified in 4.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

*(1) If your future inspection plans do not include a qualified visual examination at the next scheduled refueling outage, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

*(2) The corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.*

**RESPONSE:**

The technical basis for concluding that regulatory bases are met for Ginna Station is provided in MRP-48 [Ref. 1] and the previous inspections as described above. Corrective actions to be taken for indications will be documented in the letter committed to the NRC by December 31, 2001. If RG&E elects to not perform a visual or volumetric inspection during the March 2002 refueling outage, this will be documented in the letter, including the basis for concluding applicable regulatory requirements will continue to be met.

**(5)** *Addressees are requested to provide the following information within 30 days after plant restart following the next refueling outage:*

*a.* *a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;*

*b.* *if cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.*

**RESPONSE:**

RG&E will provide the requested information within 30 days after plant restart following the March 2002 refueling outage.

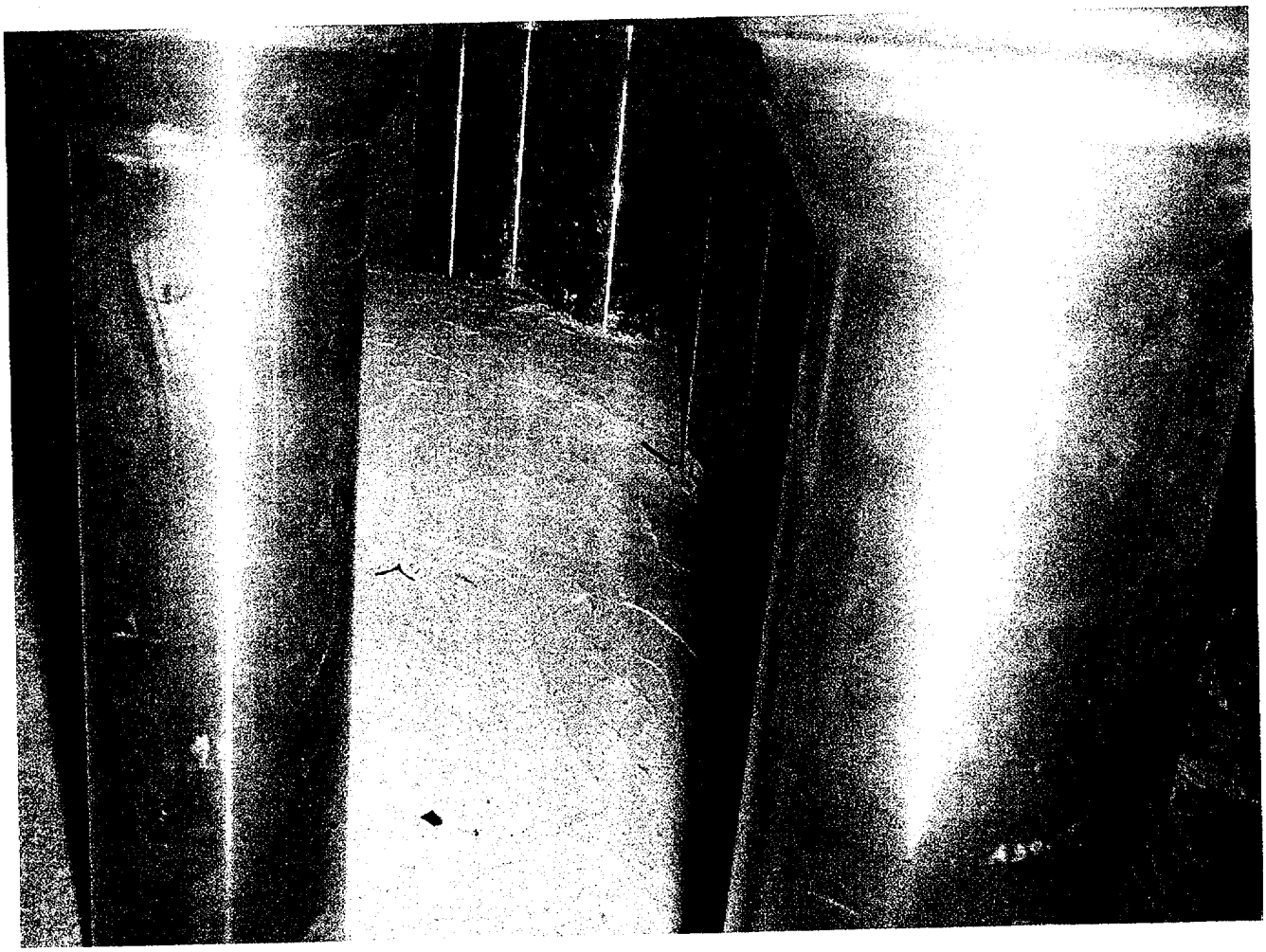
**REFERENCES:**


1. PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48), EPRI, Palo Alto, CA, 2001.

2. Ref. 12 of WCAP 15143, "USNRC Letter, A.G. Hansen to R.E. Link, "Acceptance Criteria for Control Rod Drive Mechanism Penetrations at Point Beach Nuclear Plant, Unit 1," March 9, 1994."

**ENCLOSURE 2**

**PHOTO OF GINNA STATION REACTOR HEAD INSULATION**

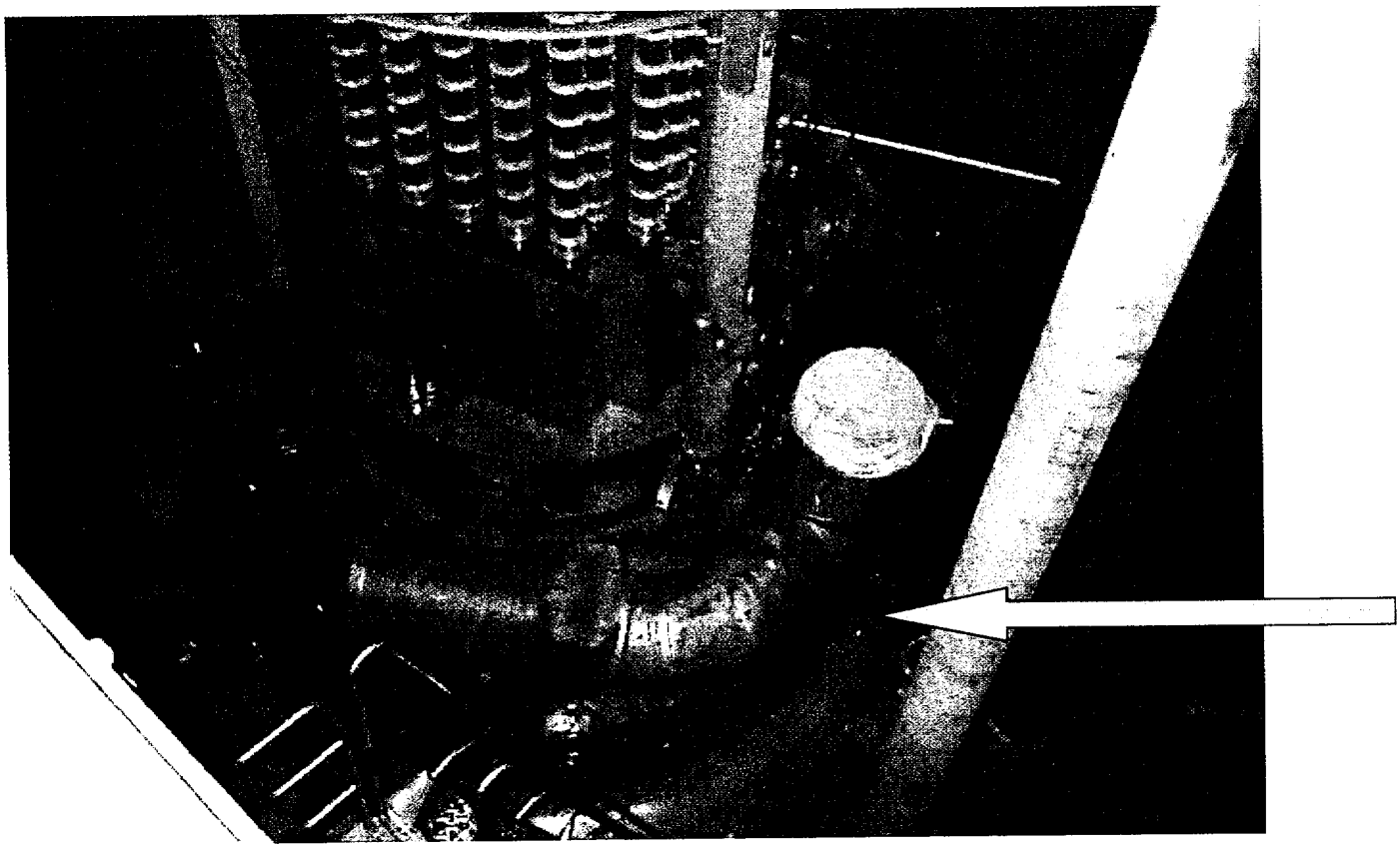


CRD.jpg

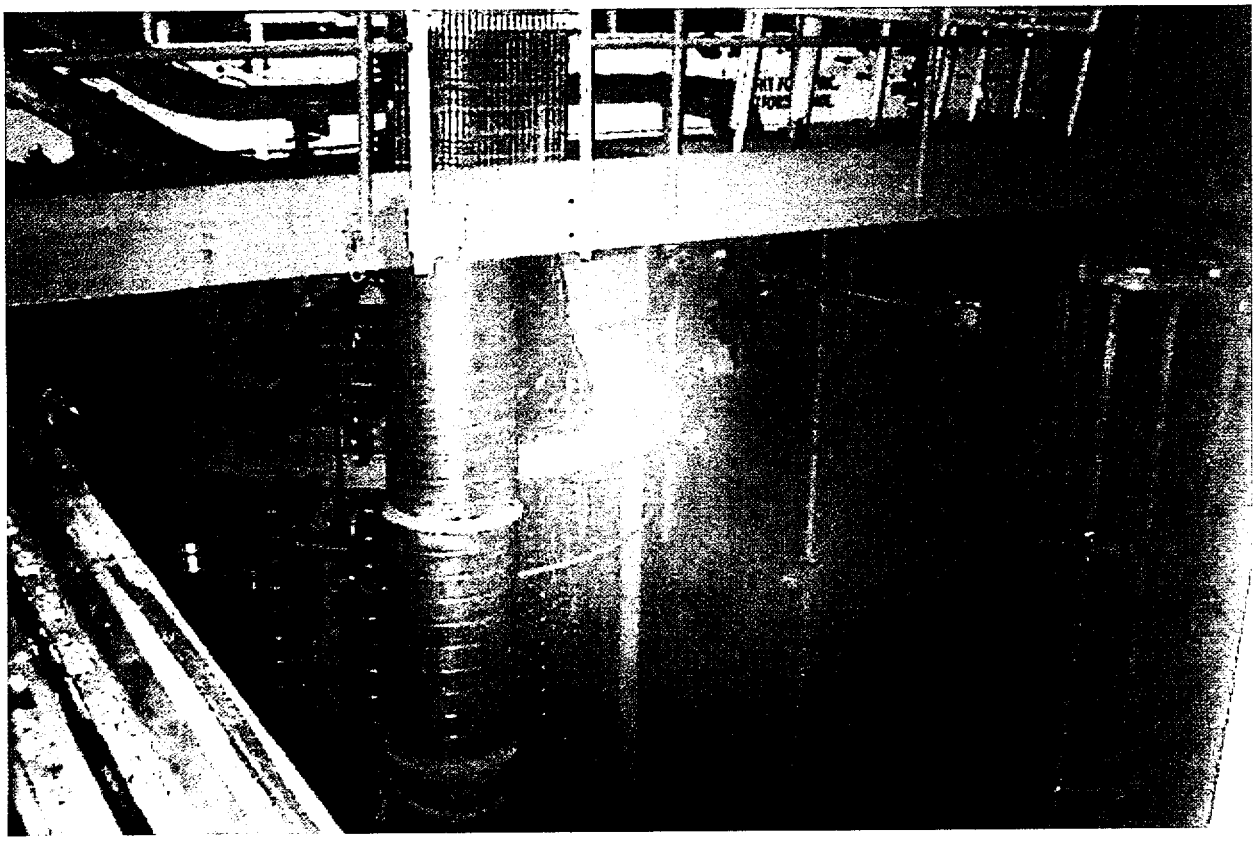
## ENCLOSURE 3

## GENERAL AREA CONFIGURATION PHOTOGRAPHS

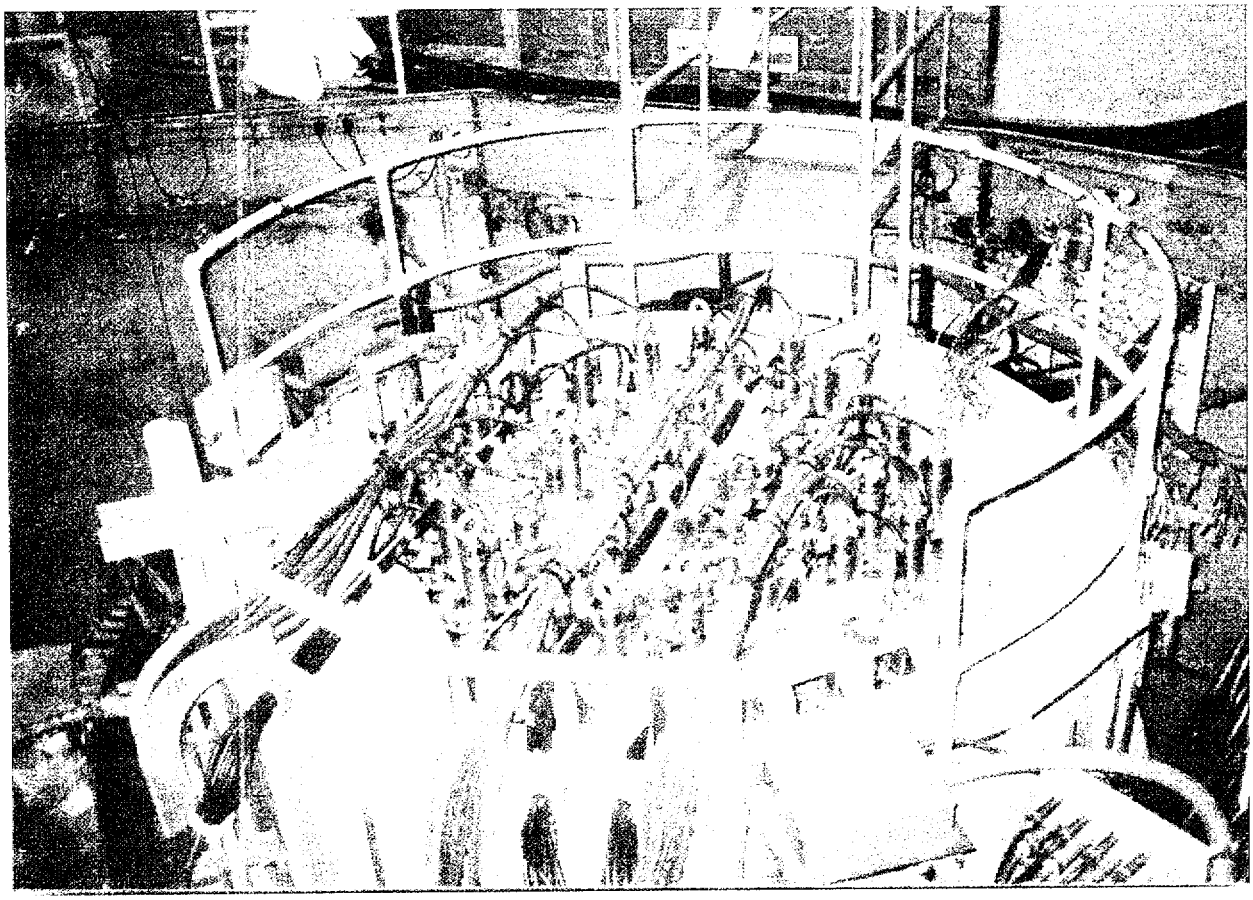
3A - Reactor Vessel Head Region in the Refuel Cavity
3B - Missile Shield Area
3C - Missile Shield Area
3D - Reactor Vessel Upper Head Seismic Restraint



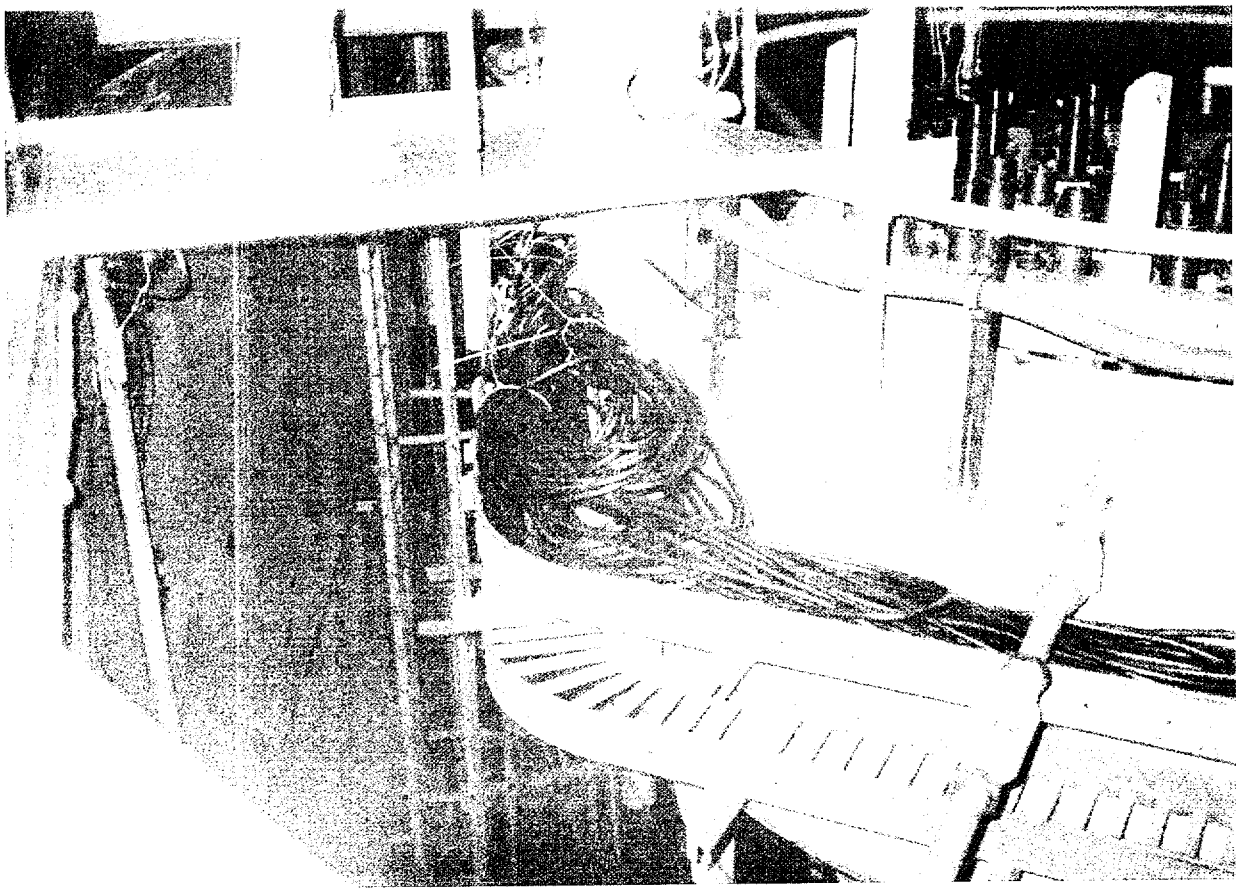
Enclosure 3A



Enclosure 3B



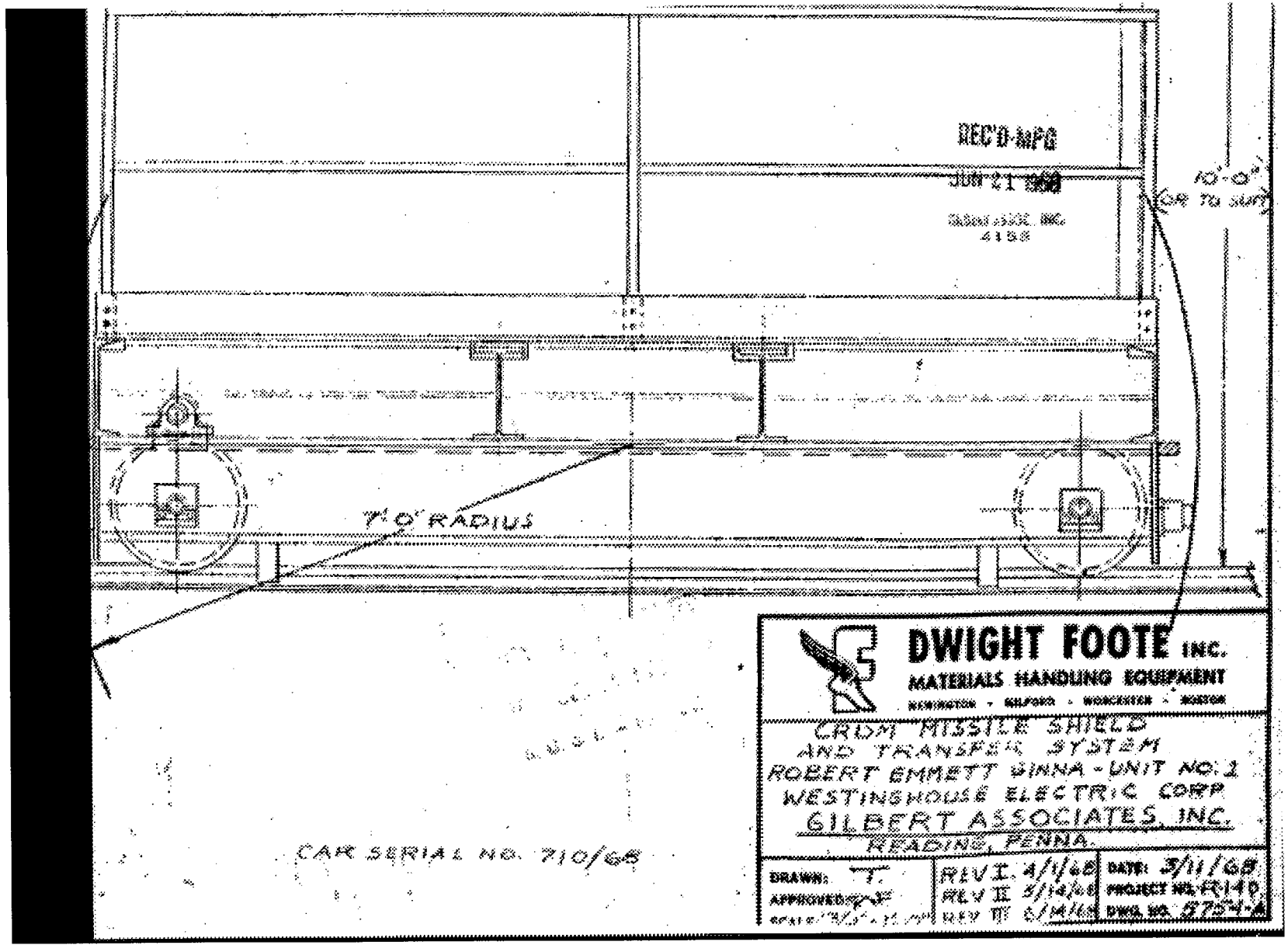
Enclosure 3C

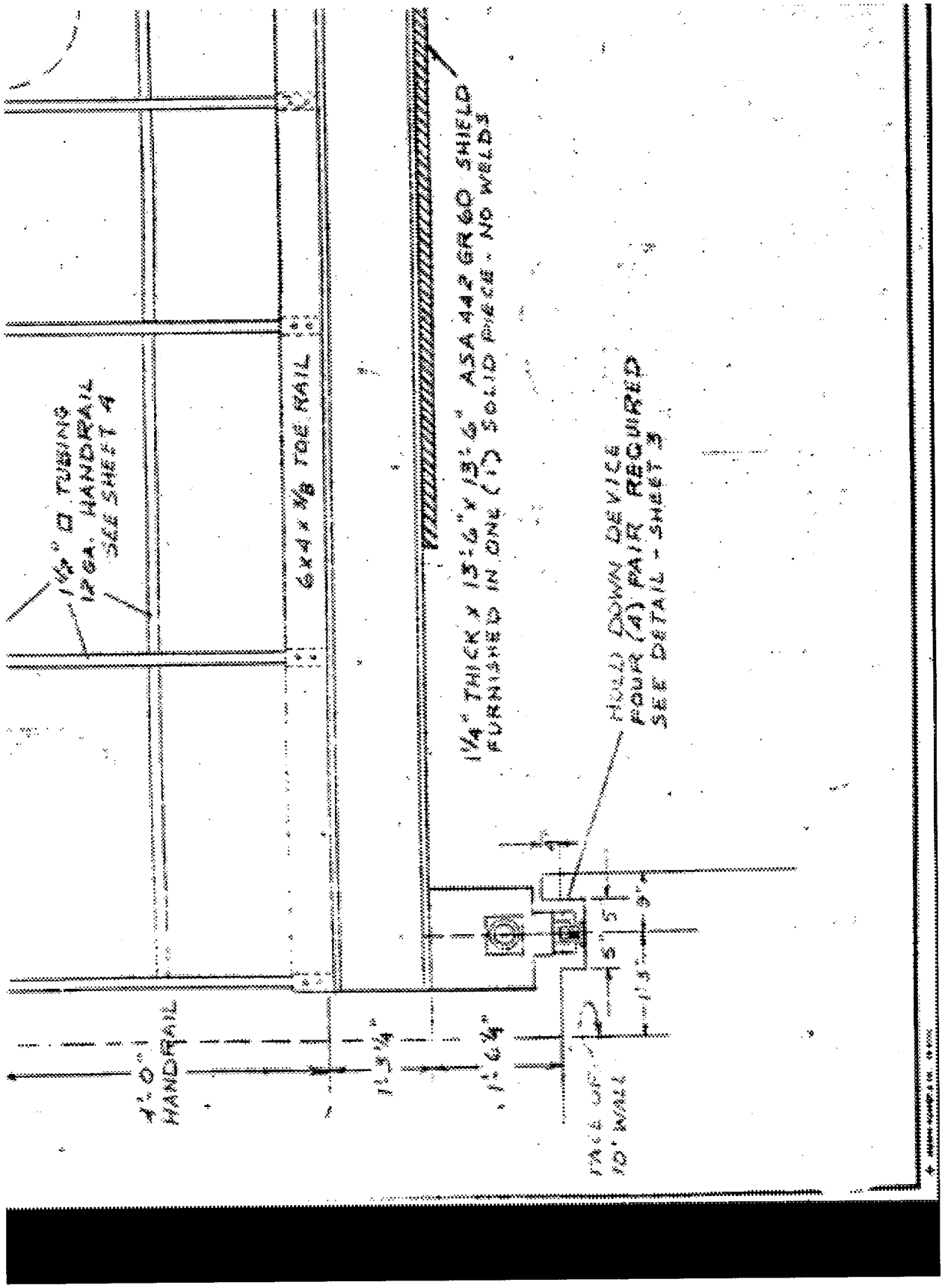


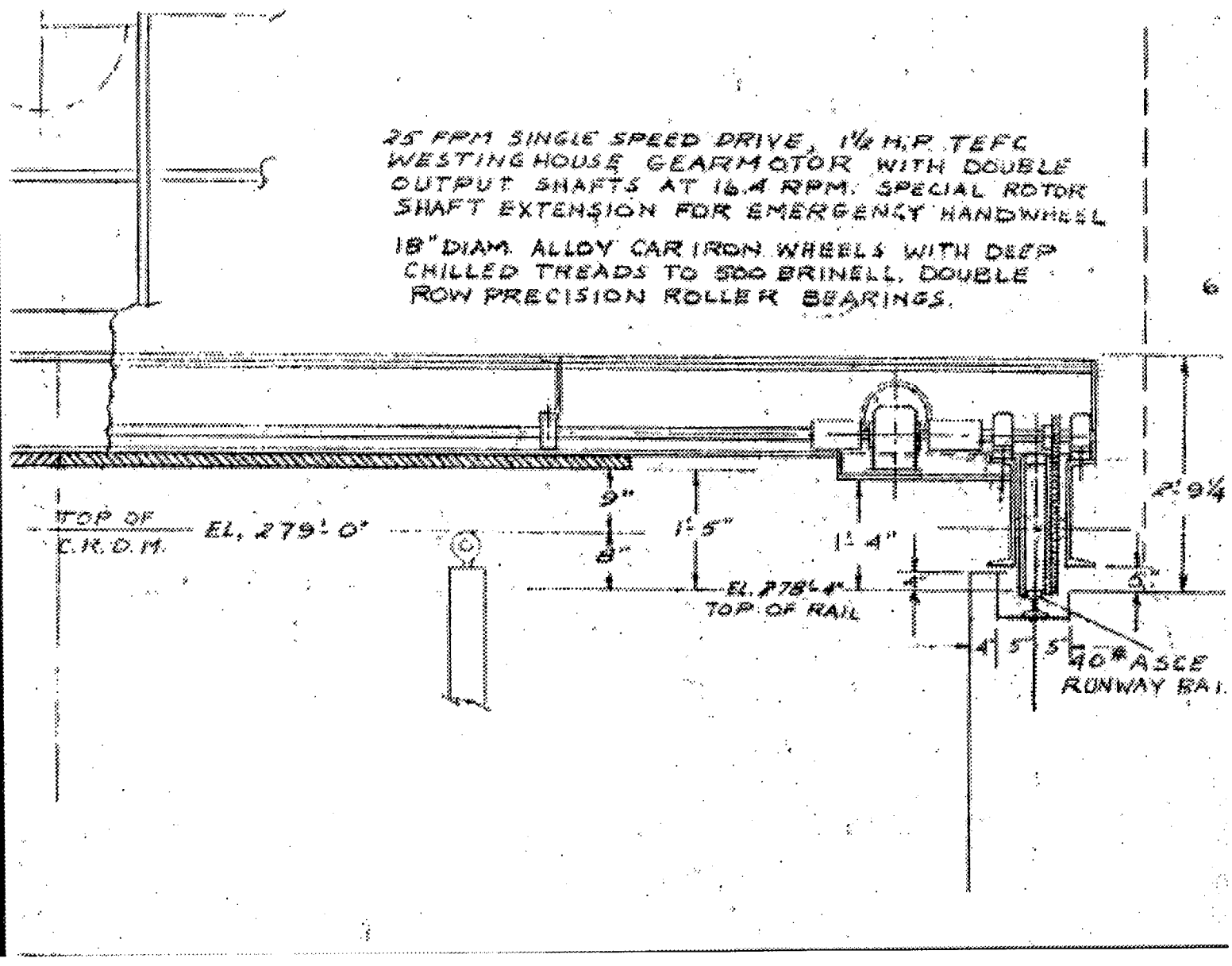
Enclosure 3D

ENCLOSURE 4

OEM DRAWING OF MISSILE SHIELD


REC'D-MPG
JUN 21
4158
10'-0"
OR TO SUIT
7'-0" RADIUS
CAR SERIAL NO. 710/68
DWIGHT FOOTE INC.
MATERIALS HANDLING EQUIPMENT
CRDM MISSILE SHIELD
AND TRANSFER SYSTEM
ROBERT EMMETT GINNA - UNIT NO. 1
WESTINGHOUSE ELECTRIC CORP
GILBERT ASSOCIATES, INC.
READING, PENNA.
DRAWN: T.
REV I 4/1/68
REV II 5/14/68
REV III 6/4/68
DATE: 3/11/68
PROJECT NO. F-14 D
DWG. NO. 8754-A


1½" O. TUBING
12 GA. HANDRAIL
SEE SHEET 4
6x4x⅜ TOE RAIL
1¼" THICK x 13'-6" x 13'-6" ASA 442 GR 60 SHIELD
FURNISHED IN ONE (1) SOLID PIECE - NO WELDS
HOLD DOWN DEVICE
FOUR (4) PAIR REQUIRED
SEE DETAIL - SHEET 3
4'-0"
HANDRAIL
12'-3¼"
1'-6¾"
5'-5"
9"
1'-8"
TACK OFF
10' WALL


25 RPM SINGLE SPEED DRIVE, 1½ H.P. TEFC
WESTINGHOUSE GEARMOTOR WITH DOUBLE
OUTPUT SHAFTS AT 16.4 RPM. SPECIAL ROTOR
SHAFT EXTENSION FOR EMERGENCY HANDWHEEL
18" DIAM. ALLOY CAR IRON WHEELS WITH DEEP
CHILLED TREADS TO 500 BRINELL. DOUBLE
ROW PRECISION ROLLER BEARINGS.
TOP OF
C.R.D.M.
EL. 279'-0"
9"
8"
1'-5"
1'-4"
EL. 278'-4"
TOP OF RAIL
2'-9¼
3¼"
4
5
5
90# ASCE
RUNWAY RAIL

ENCLOSURE 5

EC AND UT INSPECTION PROBES USED DURING 1999 INSPECTION

## Probe Types

The eddy current probe types used at Ginna station during the 1999 CRDM nozzle inspection included a blade probe (also known as a gap scanner) for the nozzles with thermal sleeves. A motorized rotating probe coil (MRPC) probe for the vent pipe, and a larger diameter MRPC probe with the identical size inspection coils for the instrumentation ports without thermal sleeves. All of these exact coil designs were qualified on the EPRI blind test samples. In addition the essential variables of the exam were also documented.

The blade probe consisted of two 4.0 mm diameter pancake probes offset by 45°. These probes were energized in both absolute and differential modes using 600 kHz, 280 kHz, and 100 kHz. frequencies. The alignment of these probes and the use of multifrequency inspection enabled the inspection probe to be sensitive to a variety of flaw geometry's with axial and circumferential orientations.

The MRPC probes used 3 coils, the first was a .080" pancake coil, and the other two were pancake coils. One of the additional pancake coils was oriented for axial flaw detection while the other pancake coil was oriented for circumferential flaw detection. These coils were also energized in both absolute and differential modes using 600 kHz, 280 kHz, and 100 kHz. frequencies. The design of these probes and the use of multifrequency inspection enabled the inspection probe to be sensitive to a variety of flaw geometry's with axial and circumferential orientations.

The blind demonstration that was performed on the EPRI mockups included the actual inspection end effectors that were used at Ginna. The nozzles were also inspected in the same design configuration as the Ginna CRDM nozzles. The EPRI mockup consisted of notches which were further subjected to the closed isostatic process (CIP) manufactured flaws which had demonstrated response equivalency to real flaws. The flaws were manufactured into multiple nozzles and covered Axial and Circumferential directions, individually isolated as well as multiple clustered flaws, flaws skewed up to 45°, flaws that exhibited branching, and closely spaced parallel flaws. The blind demonstration on both the gap scanner and the MRPC probe achieved 100% detection on all flaws. It was demonstrated that an inside diameter surface breaking flaw would be detected regardless of whether the initiation point was from the inside nozzle diameter or outside nozzle diameter.

## Ginna Station Nozzle Inspection Area

The area of interest for the Ginna inspection was the inside surface of the alloy 600 material, approximately 2" above the Control Rod Drive Mechanism (CRDM) Nozzle to head weld and 2" below the CRDM nozzle to head weld.