Duke Power®
A Duke Energy Company

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September 8, 2004

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Request for Relief Number 03-001
Reply to Request for Additional Information

Reference: Letter from NRC to Duke Energy Corporation, dated April 20, 2004

Pursuant to 10 CFR 50.4, please find attached the subject reply to the reference letter. The format of the reply is to restate the NRC question, followed by Catawba's response.

There are no regulatory commitments contained in this letter or its attachment.

If you have any questions concerning this material, please call L.J. Rudy at (803) 831-3084.

Very truly yours,

Dhiaa M. Jamil

LJR/s

Attachment

A047

xc (with attachment):

W.D. Travers, Regional Administrator
U.S. Nuclear Regulatory Commission, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

E.F.. Guthrie, Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Catawba Nuclear Station

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Mail Stop O-8 G9
Washington, D.C. 20555-0001

REQUEST FOR ADDITIONAL INFORMATION

DUKE POWER COMPANY

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

The Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's submittal dated May 22, 2003, requesting relief from performing volumetric examinations on the Catawba Nuclear Station (Catawba), Units 1 and 2, Regenerative Heat Exchangers as required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), 1989 Edition for the second 10-year interval inservice inspection (ISI) program. The NRC staff has identified the following information that is needed to enable the continuation of its review.

1. Confirm the second interval end dates for Catawba, Units 1 and 2, are June 28, 2005, and August 18, 2006, respectively.

Catawba response:

Catawba Unit 1 commercial operation began on June 29, 1985. Catawba Unit 2 commercial operation began on August 19, 1986. This translates into the second interval end dates as stated above.

2. You requested relief from Examination Category C-A requirements for head-to-shell and tubesheet-to-shell welds on Catawba, Units 1 and 2, regenerative heat exchangers. The drawings you provided also show Class 2 nozzle-to-shell welds. Confirm that these nozzle-to-shell welds are exempt from volumetric and/or surface examination requirements per IWC-1222. If they are not exempt, provide information on any dose burden associated with the examination requirements for these welds.

Additionally, typical Westinghouse designed plants have regenerative heat exchangers that are three horizontal tube and shell type vessels connected in series, stacked vertically. The drawings you provided only show one of the three vessels for each unit. Provide drawings or describe the actual configuration of the heat exchangers in their entirety, showing interconnecting piping and other appurtenances. Also provide more detailed drawings that show cross-sectional views of the head-to-shell and tubesheet-to-shell welds included in this request. The drawings should list the materials' specifications, dimensions of the components, and clearly indicate interferences for performing ultrasonic and surface examinations. Include such drawings for the aforementioned nozzle-to-vessel welds, as applicable.

Furthermore, the ASME Code Table IWC-2500, Examination Category C-A, Note 3, states: "In the case of multiple vessels of similar design, size, and service (such as steam generators and heat exchangers), the required examinations may be limited to one vessel or distributed among the vessels." Discuss Duke's interpretation of Note 3, and more specifically, the pertinence of this note to regenerative heat exchanger welds at Catawba, Units 1 and 2.

Catawba response:

ASME Section XI, Paragraph IWC-1222(b), applies to component connections nominal pipe size four inches and smaller (including nozzles, socket fittings, and other connections) in vessels, piping, pumps, valves, and other components. Chemical and Volume Control System piping welded to the Regenerative Heat Exchanger Nozzles is three inches in diameter. Therefore, these nozzle-to-shell welds are exempt from volumetric and/or surface examination requirements per IWC-1222.

See enclosed Joseph Oat and Sons, Incorporated drawings as listed below:

Drawing No. 5581, Outline Drawing 3 Shell Regenerative Heat Exchanger
Drawing No. 5582, Regenerative Heat Exchanger Unit Assembly RG-703
Drawing No. 5583, Sub Assembly + Details Regenerative Heat Exchanger RG-703

Duke Energy Corporation considers the three shell regenerative heat exchangers with the connecting piping, as shown on Joseph Oat and Sons, Incorporated outline drawing No. 5581, as one vessel; therefore, Note 3 does not apply. The regenerative heat exchanger is shown on the Catawba Nuclear Station Chemical and Volume Control System Flow Diagram Drawings as one piece of equipment.

3. You stated that flow induced vibration in letdown system piping had been observed in the past at Catawba, Units 1 and 2, and noted that vibrational loads emanating from the

letdown orifices were attenuated by the (regenerative) heat exchanger configuration and its distance from the vibration source. You also indicated that you made modifications to reduce vibration in the letdown piping. The NRC staff acknowledges that, when compared to other pressurized water reactor (PWR) systems, most fatigue failures have occurred in chemical and volume control system (CVCS) piping, mainly caused by vibrational fatigue in either letdown or charging piping. However, most recently, a vibration fatigue failure was reported at the regenerative heat exchanger letdown nozzle outlet weld due to flow-induced vibration from positive displacement charging pump operation. In many cases, vibrational fatigue damage may occur during specific operating configurations. For example, in the aforementioned failure, vibrational loads were highest when only a single charging pump was in operation. Since single charging pump operation occurred infrequently, and since the piping was inside containment and inaccessible during normal operation, this condition was never identified nor observed during system walkdowns or Code-required system leakage tests. The other principal source of high vibration in CVCS piping has been from the letdown orifices, which is consistent with the experience at Catawba.

Therefore, provide additional details related to the vibration problems noted in the Catawba letdown lines and subsequent plant modifications. Also, describe the modifications' impact on measured vibration loads. Describe the operating practices (e.g., plant conditions, system configurations, and operating history, etc.) for the reciprocating positive displacement pumps at Catawba, Units 1 and 2. Identify peak velocities in letdown and charging piping between the regenerative heat exchanger nozzles (inlet and outlet) and the first fixed or pinned support, for all letdown orifice and charging pump (centrifugal and reciprocating) operating combinations. Confirm that each peak velocity is less then the allowable velocity criterion specified in ASME standard OM-S/G-1990, "Requirements for Pre-operational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems."

Catawba response:

Catawba does not operate positive displacement charging pumps on either unit. Vibration monitoring of piping between the regenerative heat exchanger nozzles and the first fixed or pinned support for all letdown orifice and charging pump combinations would result in significant personnel dose. Therefore, the requested section of the

piping has not been monitored with centrifugal pumps operating.

Additional details related to vibration problems noted in the Catawba letdown lines and subsequent plant modifications are discussed below. Any modifications' impact on measured vibration loads is also discussed.

Unit 1

Nuclear Station Modification NSM CN-11343 was implemented in the mid 1990s to address socket weld failures resulting from high vibration levels in the chemical and volume control letdown lines. The NSM replaced the letdown orifices with a more advanced design, replaced the cavitation trim in variable orifice NV-849, and deleted socket welds in the letdown line that were susceptible to failure from high vibration levels. Post-modification testing was performed on January 3, 1998 to measure vibration in the Unit 1 letdown lines and set the travel stop on NV-849. The testing showed that the modification had successfully reduced vibration levels in the letdown lines. However, three of the sixty recorded data points exceeded the ASME OM-S/G 1990 (a.k.a. OM-3) screening criteria for peak velocity of vibration of stainless steel pipe. Namely, maximum peak velocity in the vertical direction at valve 1NV-13A was 1.105 ips (45 gpm + variable orifice in service) at 115 gpm letdown flow and 0.956 ips at the same location but with 75 gpm and variable orifice in service. The third point was at a butt-welded flow orifice component 1NVFE5970 and not of concern. Acceleration was only 0.74 ips.

An engineering evaluation was performed for the 1NV-13A vibration reading using EPRI velocity-based screening criteria (reference EPRI TR-104534-V1, V2, and V3, "EPRI Fatigue Management Handbook") and it was found to be acceptable.

A second chemical and volume control letdown line vibration test was conducted on May 21, 1999. The results of this second test indicated acceptable levels of vibration when considering the OM-3 criteria of 0.69 ips with the following exception: velocity of 1.1 ips measured at 1NVFE5970 and 1NV-13A while the 75 gpm orifice was in service. This was judged acceptable based on the EPRI velocity-based screening criteria. For the variable orifice case (100 gpm flow), no valid data could be obtained at 1NV-10A and 1NV-11A.

Therefore, operation at 100 gpm through the variable orifice was not recommended.

During the End of Cycle 13 Refueling Outage in 2002, NSM CN-11416 installed a newly-designed variable orifice 1NV-849. Old 1NV-849 was replaced due to degradation of internals caused by cavitation on its Unit 2 counterpart, 2NV-849. Post-modification testing on May 26-27, 2002 of the letdown lines was conducted following the installation of the new 1NV-849. Three flow regimes were selected for this monitoring: 1) flow of 110 gpm split between the 75 gpm orifice and the variable orifice, 2) 110 gpm through the variable orifice only, and 3) flow of 75 gpm through the variable orifice only, which would be the normal alignment for future operation. Monitoring was performed at socket welded valves 1NV-10A, 1NV-11A, 1NV-13A, variable orifice 1NV-849, and orifices 1NVFE5950 and 1NVFE5970. All results were within the EPRI velocity-based acceptance criteria.

On July 17, 2003, additional vibration data was collected on the Unit 1 letdown lines to determine vibration levels in piping while operating at a reduced backpressure of 150 psi (to minimize leakage past the relief valve 1NV-14) with flow at 70 gpm through the variable orifice. Valves 1NV-10A, 1NV-11A, 1NV-13A, and variable orifice 1NV-849 velocities were monitored and fell within the OM-3 and EPRI velocity-based criteria. Additionally, cantilever vent valves 1NV-901, 1NV-920, and 1NV-836 were monitored and found to be acceptable using EPRI acceleration-based criteria. These vent valves are well downstream of the letdown orifices and were monitored in response to a Unit 2 leak at high point vent valve 2NV-950 that developed in February, 2003 (see Unit 2 discussion).

Unit 2

NSM CN-21343 is the sister modification to Unit 1 NSM CN-11343 described above. Post-modification vibration testing was performed on November 15, 1995 and again on March 25, 2003 following implementation of NSM CN-21416 to replace 2NV-849 during the End of Cycle 12 Refueling Outage (the sister modification to Unit 1 NSM CN-11416 described above). Data was collected at 2NV-10A, 2NV-11A, 2NV-12, 2NV-13A, 2NV-849, and 2NV-950 and was found to be acceptable using EPRI velocity or acceleration-based criteria, as appropriate. However, valve 2NV-950 was only considered

marginally acceptable using the EPRI criteria and is discussed below.

Just prior to replacement of 2NV-849 during the End of Cycle 12 Refueling Outage, a leak developed in the socket weld below vent valve 2NV-950 in February 2003. A root cause evaluation indicated that the leak was most likely attributable to vibration fatigue of the weld. The weld repair consisted of a socket weld as before, but with a 2:1 taper for increased resistance against vibration fatigue. Following the repair, vibration measurements were collected on February 25, 2003. Maximum peak accelerations of 30 g (SRSS of three directions) were measured at the pipe cap above 2NV-950 with 110 gpm flow through 2NV-849. Following replacement of 2NV-849 in March 2003, the maximum peak acceleration (SRSS of three directions) at the 2NV-950 pipe cap was measured at 10 g (and 4.8 g near the valve center of gravity). Therefore, the replacement of 2NV-849 resulted in a significant reduction in the vibration at this valve location. Still, the vibration data indicated bending stresses approximately equal to the recommended endurance limit of 10,880 psi for stainless steel. Thus a long-term solution for 2NV-950 is a redesign to eliminate the socket weld altogether (there is a current corrective action program activity to accomplish this). A separate corrective action will monitor vibration in cantilever vents and drains further downstream of 2NV-950 to the outboard containment isolation valve.

4. You indicated that average radiation levels near the regenerative heat exchangers at Catawba, Units 1 and 2, are approximately 0.7rem/hr. In order to attain these dose rates, a peroxide induced crud burst and subsequent water flush of the letdown lines and heat exchangers is performed each outage. Provide additional information describing how this procedure is performed, including chemical species present, flush path, flush time, component temperatures, and plant components (pumps) used to perform the flush. Assess the impact of this flushing operation on the continued structural integrity of the subject heat exchanger welds and confirm whether these existing crud control measures will continue to be performed.

Catawba response:

Prior to the peroxide induced crud burst of the reactor coolant system, the normal letdown line, including the regenerative heat exchanger shell side, is isolated.

Flush of the shell side of the regenerative heat exchanger is performed after normal letdown has been taken out of service if the flush is required to reduce dose rates. The regenerative heat exchanger shell side is isolated after letdown from the Residual Heat Removal System is placed in service, which typically takes place between 200°F and 230°F. Once letdown is isolated, the temperature of the heat exchanger and connecting piping will cool to charging fluid temperature, which is typically less than 105°F. The flush is performed with water from the Reactor Makeup Water Pumps which have the RMWST (Reactor Makeup Water Storage Tank) as the suction source. The water is introduced into the letdown line just downstream of the regenerative heat exchanger and is flushed through the heat exchanger and out of a drain in the letdown piping on the inlet side of the heat exchanger. Temperature of the RMWST water typically ranges from 40°F to 105°F. Dissolved oxygen in the RMWST is routinely sampled, with the results typically in the range of 0.4 ppm to 1 ppm. The flush is performed for approximately 45 minutes or until Radiation Protection directs that the flush be stopped. The impact of the flush on the structural integrity of the subject heat exchanger welds cannot be determined. The flush is only performed if dose rates in the vicinity of the letdown line and/or heat exchanger indicate a need to perform the flush.

5. Duke stated that oxygen levels in the primary system are strictly limited, thereby reducing the susceptibility to intergranular stress corrosion cracking (IGSCC), and noted that the nuclear power industry's operating experience suggests that the regenerative heat exchanger materials (welds and base materials) are not susceptible to significant corrosion IGSCC in primary water environments. The NRC staff agrees that during normal operation, primary water chemistry conditions are such that oxygen concentrations are expected to be very low. However, industry service experience has reported several stress corrosion cracking failures in PWR austenitic stainless steel (Type 304/316) piping systems. For Catawba, Units 1 and 2, regenerative heat exchangers, identify the ASME material specifications including mechanical and chemical properties. Identify durations and plant conditions when the regenerative heat exchangers and connecting piping are exposed to oxygen or oxidizing species and the temperatures are greater than 150 degrees F, regardless of the plant operation mode.

Catawba response:

The materials associated with the heat exchanger are specified below:

Heads:	SA240	TP304L
Nozzles:	SA479	TP304
Shell:	SA351	Grade CF8
Interconnecting pipe:	SA312	TP316

The regenerative heat exchangers and connecting piping are not routinely exposed to oxygen or oxidizing species when the temperatures are greater than 150°F except during startup. During shutdown, the regenerative heat exchanger shell side is isolated after letdown from the Residual Heat Removal System is placed in service. Once the shell side is isolated, the temperature of the heat exchanger will drop to the temperature of the charging fluid, which is typically less than 105°F. The introduction of peroxide for the crud burst takes place after the temperature of the regenerative heat exchanger is less than 150°F.

During startup, dissolved oxygen is required to be within specification prior to Mode 4 (200°F). This leaves the period between 150°F and Mode 4 where the dissolved oxygen may not be within normal specifications. The past four startups for Unit 1 were reviewed and the time between 150°F and 200°F ranged from 15 hours to 38 hours, with an average of approximately 28 hours.

6. Confirm that all Category C-A welds identified in Request for Relief 03-001 have been volumetrically inspected at least once during fabrication, pre-service inspection, or ISI. Describe the results of these examinations, and identify whether weld repairs have been performed on any of the subject welds.

Catawba response:

All Category C-A welds identified in Request for Relief 03-001 were volumetrically inspected by radiography during vessel fabrication. A weld repair was performed by the vessel manufacturer on the Catawba Unit 1 vessel, shell number 2, girth weld number 2. The repair was limited to one area contained within one four-inch RT film interval (4-5). In addition, a weld repair was performed on the Catawba

Unit 2 vessel, shell number 3, girth weld number 1. The repair was limited to one area contained within two four-inch film intervals (4-5 and 5-6). The weld repair areas were re-examined by radiography and found to be acceptable. The remaining welds on the Catawba Unit 1 and 2 vessels were found to be radiographically acceptable during the fabrication process without performing weld repair activities.

7. You stated that a reactor coolant leak detection system is in place to detect any variation in reactor water inventory, including water levels present in both the shell and tube side of the regenerative heat exchangers. You further state that any (regenerative heat exchanger) weld failure would be detected by this leak detection system and that procedures and automatic system actions are in place to ensure that the heat exchangers would be isolated. Provide additional information describing the reactor coolant leak detection system, leakage measurement and prediction techniques, leakage monitoring frequencies, redundancy, and regenerative heat exchanger leak rate sensitivity. Identify the [regenerative heat exchanger] leakage flaw size (length and crack opening displacement) that will assure detection by the reactor coolant leakage detection system. This flaw size should be sufficient to assure that leakage is detected with a margin for uncertainties consistent with NRC leak-before-break evaluation procedures and identify the margin to critical (unstable) crack size. Also, describe the procedures and automatic system actions that are in place to isolate the regenerative heat exchangers.

Catawba response:

Plant Technical Specifications dictate that a reactor coolant system water inventory balance be performed on a regular basis (i.e., at least once every three days). The normal operating practice is to perform this computer based program on a daily frequency and/or whenever the operators suspect any abnormal changes to other leakage detection systems. Plant Technical Specifications require that system leakage from unidentified sources be maintained below 1 gpm; however, plant operating procedures establish an administrative limit of 0.15 gpm, above which the source of the leakage must be investigated. Leakage as a result of a failed weld discussed in this section would show up as unidentified leakage and would be subject to the 0.15 gpm administrative limit.

The water inventory balance provides repeatable results less than the 0.15 gpm administrative limit; however, an evaluation of sensitivity below this leak rate level has not been performed.

Other leakage detection systems available to the operator and dictated per plant Technical Specifications are:

- Containment Atmosphere Gaseous and Particulate Radioactivity Monitoring System, which detects airborne radiological activity

- Containment Floor and Equipment Sump Level and Flow Monitoring Subsystem, where unidentified accumulated water on the containment floor is monitored and evaluated as sump level changes

- Containment Ventilation Unit Condensate Drain Tank Level Monitoring Subsystem, which collects and measures as unidentified leakage the moisture removed from the containment atmosphere

Additionally, other indicators are also available to the operator that a leak exists or may be developing:

- Containment Atmosphere Iodine Monitor

- Charging/letdown system mismatches

- Containment humidity

Experience has shown the combination of the above indicators to be reliable in identifying small leaks. An example is the leak on a weld at vent valve 2NV-950. Indications of this leak were:

- Small increased input into the Unit 2 Ventilation Unit Condensate Drain Tank that began on February 7, 2003. Previous input to the tank was essentially zero since shortly after the last Unit 2 refueling outage. The input rate to the tank was approximately 0.04 gpm.

- There was a slight increase in the Containment Floor and Equipment Sump input (approximately a 0.015 gpm increase in average combined sump input on February 8, 2003).

- There was a steady increase in upper and lower containment humidity over the week (the first noted increase was recorded on February 5, 2003 at 18:19:55).

- There was no noticeable change in Reactor Coolant System operational leakage (as measured nightly).

Catawba has not performed a leak-before-break type evaluation for the regenerative heat exchangers. Neither the leakage flaw size nor the critical crack size has been determined for these components. In order to support this type of analysis, the following steps would be required:

1. A leakage crack size is established based on a factor of ten times the leakage detection system capability.
2. A critical crack size is determined at the location of interest based on material, geometric properties, and applied loads.
3. The leakage crack size is compared with the critical crack size. The leakage crack size must be less than 0.5 times the critical crack size.
4. The critical crack size is determined based on an increased loading of 1.4 (normal plus safe shutdown earthquake) loads.
5. The leakage crack size is compared with the critical crack size. The leakage crack size must be less than the critical crack size.

The above steps would have to be performed for each weld location on the regenerative heat exchanger. It is likely that a leak-before-break type evaluation would show that the leakage crack would not produce the required leakage flow for identification by the leakage detection capability based on the small pipe size within the heat exchanger. A cost estimate for this leak-before-break type analysis is approximately $50,000 for the regenerative heat exchangers based on other recent analytical fracture mechanics evaluations.

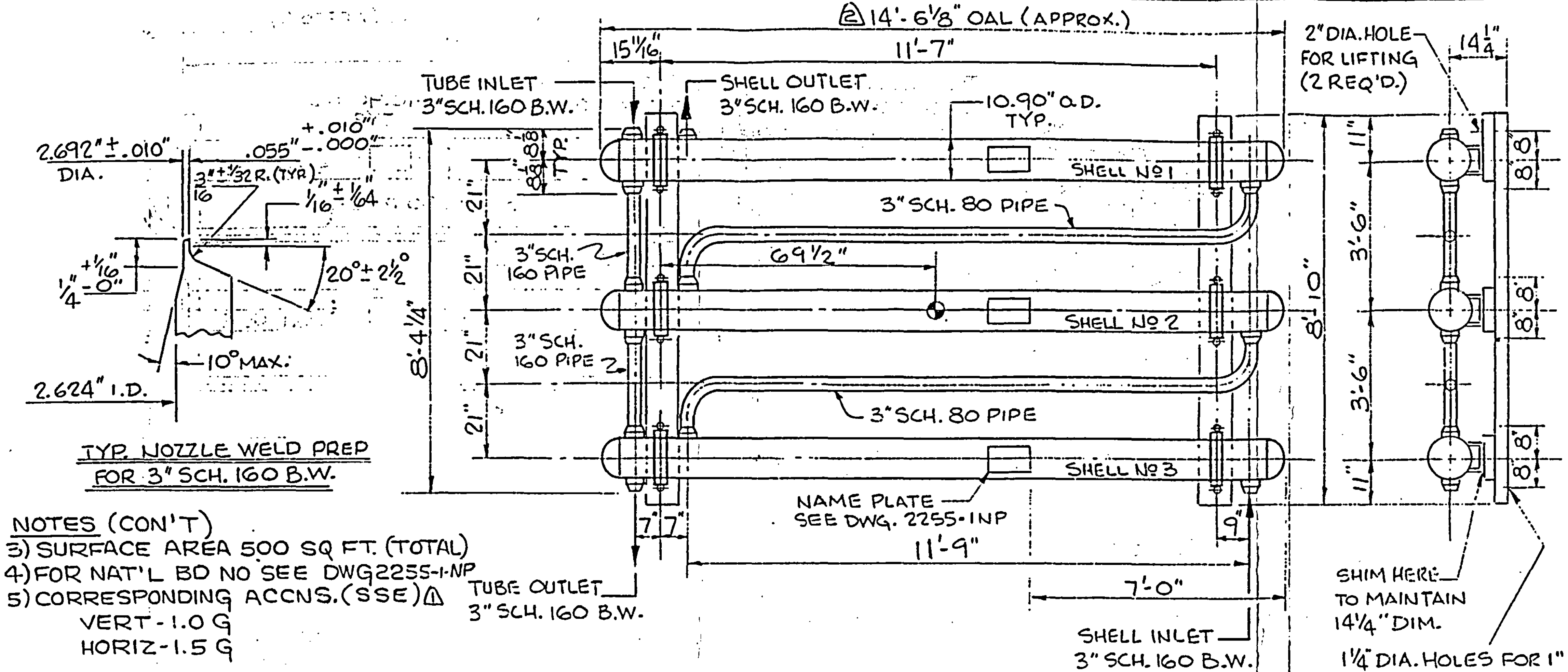
The regenerative heat exchanger is isolable from the Reactor Coolant System by valves either operated from the control room or by valves that receive automatic closure signals. The shell side of the heat exchanger is isolable from the Reactor Coolant System by two fail closed, air operated gate valves in series. These valves are provided a safety signal to automatically close on a pressurizer low level, which would be present with a significant leak from a regenerative heat exchanger shell-to-head or shell-to-tubesheet weld failure. The tube side is isolable from the high pressure charging system by two motor operated gate valves in series, which are controlled from the control room and/or which automatically close on a safety injection signal. A safety injection signal would occur with a significant heat exchanger weld leak.

Procedure AP/1(2)/A/5500/010, Reactor Coolant Leak, contains steps to respond to a Reactor Coolant System leak and includes steps to isolate letdown, utilizing the valves upstream and downstream of the regenerative heat exchanger, if the letdown line is determined to not be intact.

8. If Request for Relief 03-001 were approved, the number of Code-required Category C-A weld examinations at Catawba, Units 1 and 2, will be significantly reduced (from 26 to 14 welds for Unit 1 and from 29 to 17 welds for Unit 2). You have requested relief from the Code requirement to complete 100 percent of these Category C-A welds by the end of the current interval (Table IWC-2412-1). However, if your proposal is approved, the population of Category C-A welds available for volumetric or surface examination will, in effect, be reduced by the number of welds included in the request. Therefore, relief from IWC-2412-1 is not required. Confirm that all other Category C-A welds on all Class 2 vessels are being examined in accordance with Code requirements.

Catawba response:

All other Category C-A welds on Class 2 vessels have been reviewed and confirmed to be examined in accordance with Code requirements.



NOTES (CON'T)
3) SURFACE AREA 500 SQ FT. (TOTAL)
4) FOR NAT'L BD NO SEE DWG 2255-1-NP
5) CORRESPONDING ACCNS. (SSE) 1
VERT - 1.0 G
HORIZ - 1.5 G

ASME SECT. III CL-2, 1974 1

1 VOLUME SHELL SIDE - 12.11 CU. FT.
VOLUME TUBE SIDE - 4.08 CU. FT.

(W) P.O. No 546-CAZ-214802-BP + SPIN No DCP-CSAHRG - OAT JOB No J-2251-1A
(W) P.O. No 546-CAZ-214804-BP + SPIN No DDP-CSAHRG - OAT JOB No J-2251-1C 2

(W) ITEM No. RG 703 | OAT DWGS. - 5582-5583-5584

NOTES:
1 1. WEIGHTS (3 SHELLS + PIPING)
EMPTY - 6600# FLOODED - 7500#
1 2. GROSS SUPPORT REACTIONS
LEFT END CHANNEL VERT/HORIZ NORM - 10190#/15675#
RIGHT END CHANNEL VERT/HORIZ NORM - 18210#/8975#
HORIZ - AXIAL REACTION - 24650#

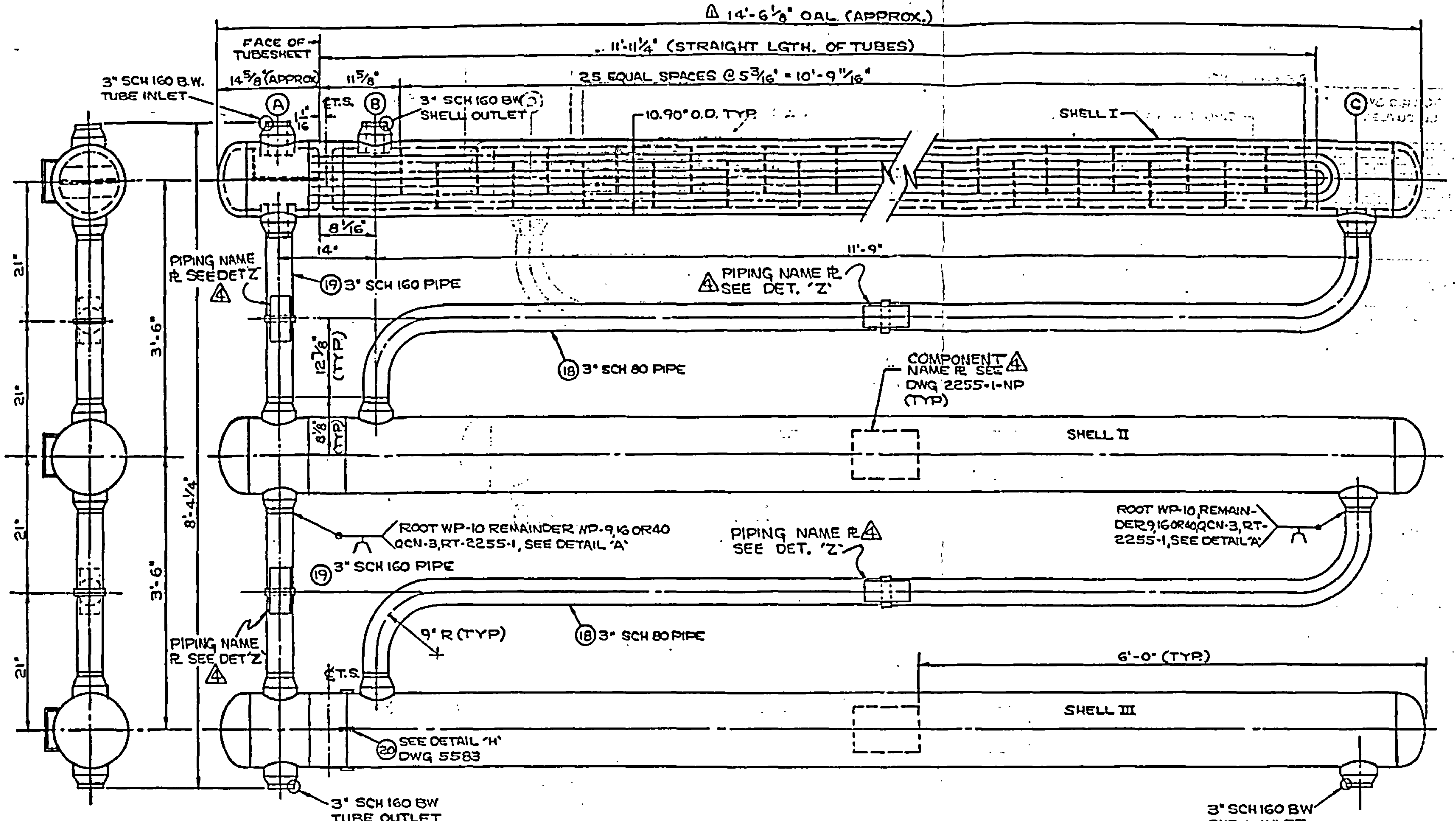
DATE	REVISION	BY
8-25-75	2 REVISED JOB & P.O. NO. FOR DDP PLANT	FMcA
1-17-75	REVISED YEAR OF CODE O A L WAS 14'-5⅛" 1 REVISED PER PAR-01 DATED 9-30-74	FMcA
7-25-74	A ISSUED FOR APPROVAL	FMcA

JOSEPH OAT & SONS, INCORPORATED
CHEMICAL ENGINEERS/DESIGNERS/FABRICATORS
ESTABLISHED 1788
CAMDEN, NEW JERSEY 08104

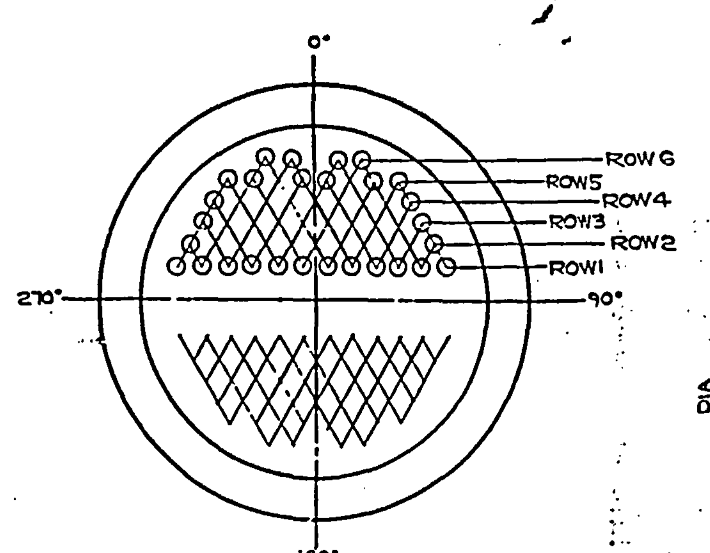
OUTLINE DRAWING 3 SHELL REGENERATIVE HEAT EXCHANGER	WESTINGHOUSE ELEC. CORP. NUCLEAR ENERGY SYSTEMS PITTSBURGH, PA.
PURCHASE ORDER NO. SEE ABOVE	NO. REQ'D. TWO

JOB NO. 2	DRAWN: R. Holmes	DRAWING NO.	REV.
J-2255-1 A/C	DATE: 7-23-74 SCALE: NONE	5581	2

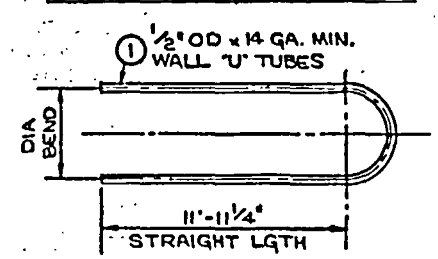
DDE 00640959



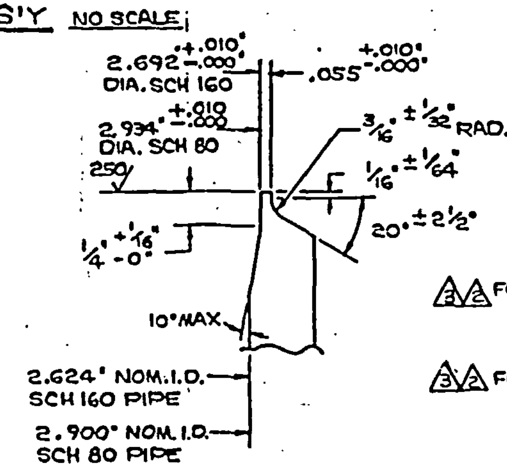
ELEV. ~ UNIT ASS'Y NO SCALE



TUBES REQ'D FOR ONE SHELL				
ITEM NO	ROW	NO REQ'D	LENGTH PER TUBE	DIA BEND
1A	1	12	24'-3 1/4"	1 3/4"
1B	2	11	24'-5 1/4"	2 21/32"
1C	3	10	24'-6 3/4"	3 29/32"
1D	4	9	24'-8 1/2"	5"
1E	5	8	24'-10"	6 3/32"
1F	6	4	25'-0"	7 5/32"



TUBE BUNDLE DATA NO SCALE

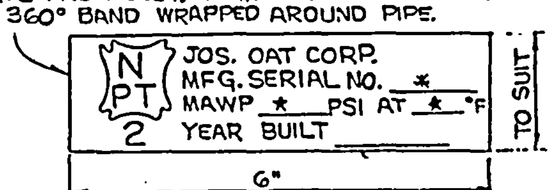


DETAIL 'A' ~ WELD PREP

DESIGN DATA	SHELL SIDE	TUBE SIDE
DESIGN & FABRICATION △1	ASME SECT. III CL 2, 1974	
DESIGN PRESSURE	2485 PSI	2735 PSI
DESIGN TEMPERATURE	650°F	650°F
CORROSION ALLOWANCE	NONE	.005" ON TUBES
RADIOGRAPH	NC-5000 (100%)	NC-5000 (100%)
FINAL TEST PRESSURE △1	3728 PSI	4103 PSI
FINAL TEST TEMPERATURE	AMBIENT	AMBIENT
MATERIAL		
SHELL	SA-351-CF8	SA-351-CF8
HEADS △1	SA-240-304L	SA-240-304L
NOZZLE NECKS △1	SA-479-304	SA-479-304
NOZZLE FLANGES		
TIE RODS	SA-479-304	
BAFFLES	SA-240-304	
SPACERS	SA-312-304	
NOZZLE REINFORCEMENT		
PASS PARTITION		SA-240-304
INTERCONNECTING PIPE	SA-312-316	SA-312-304
TUBES		SA-213-304
TUBESHEET		SA-182F-304
GASKETS		
SUPPORTS	SEE DWG.-5584	
NUTS (INTERNAL)	SA-194-8BM	
NUTS (EXTERNAL)	SA-194-2H	SA-194-2H
BOLTS	SA-193-B7	SA-193-B7

WEIGHT: EMPTY: 2500 # FULL OF WATER: 9500 #
HEAT TRANSFER SURFACE: 500 SQ. FT. (TOTAL)
STAMPING: ASME CODE SECT. III CL 2
INSPECTION: CAT, WESTINGHOUSE & UNDERWRITER

(4) S/S-304 INTERCONNECTING NAME ℞'S REQ'D LOCATE FROM ELEV. VIEW. TACK WELD TO S/S-304 360° BAND WRAPPED AROUND PIPE.



*- MARK (1) EA J-2255-1 C, D & E.
*- SEE DATA BLOCK ABOVE

△4 DET. 'Z' ~ INTER. PIPING NAME ℞
(FOR J-2255-1C, D & E ONLY)

8-25-76	△4 ADDED INTER. PIPING NAME ℞'S FOR J-2255-1 C D & E ONLY	F.M.A.	
9-25-75	△3 REVISED OAT JOB NOS. FOR DOP & NEU PLANTS. ADDED NEW P.O. NOS.	F.McA	
8-28-75	△2 ADDED NEU, LLP & LMP PLANTS (J-2255-1C, D & E)	F.M.A.	
1-10-75	△1 NOZZLES WERE SA-182F304. REVISED O.A.L., YEAR OF CODE APPLICABILTY, TEST PRESSURE & HD. MAT'L	F.M.A.	
10-28-74	△0 ISSUED FOR APPROVAL	F.M.A.	
DATE	△ REVISION	BY	APP'D

REFERENCE DWGS:
△3△2 FOR J-2255-1 A/C ONLY: OUTLINE DWG: 5581; SUB-ASS'Y DWG: 5583; SUPP'T BRKT. DWG: 5584; NAME ℞ DWG: 2255-1-NP
△3△2 FOR J-2255-1B, D & E ONLY: OUTLINE DWG: 5728; NAME ℞ DWG: 2255-1-BRKT

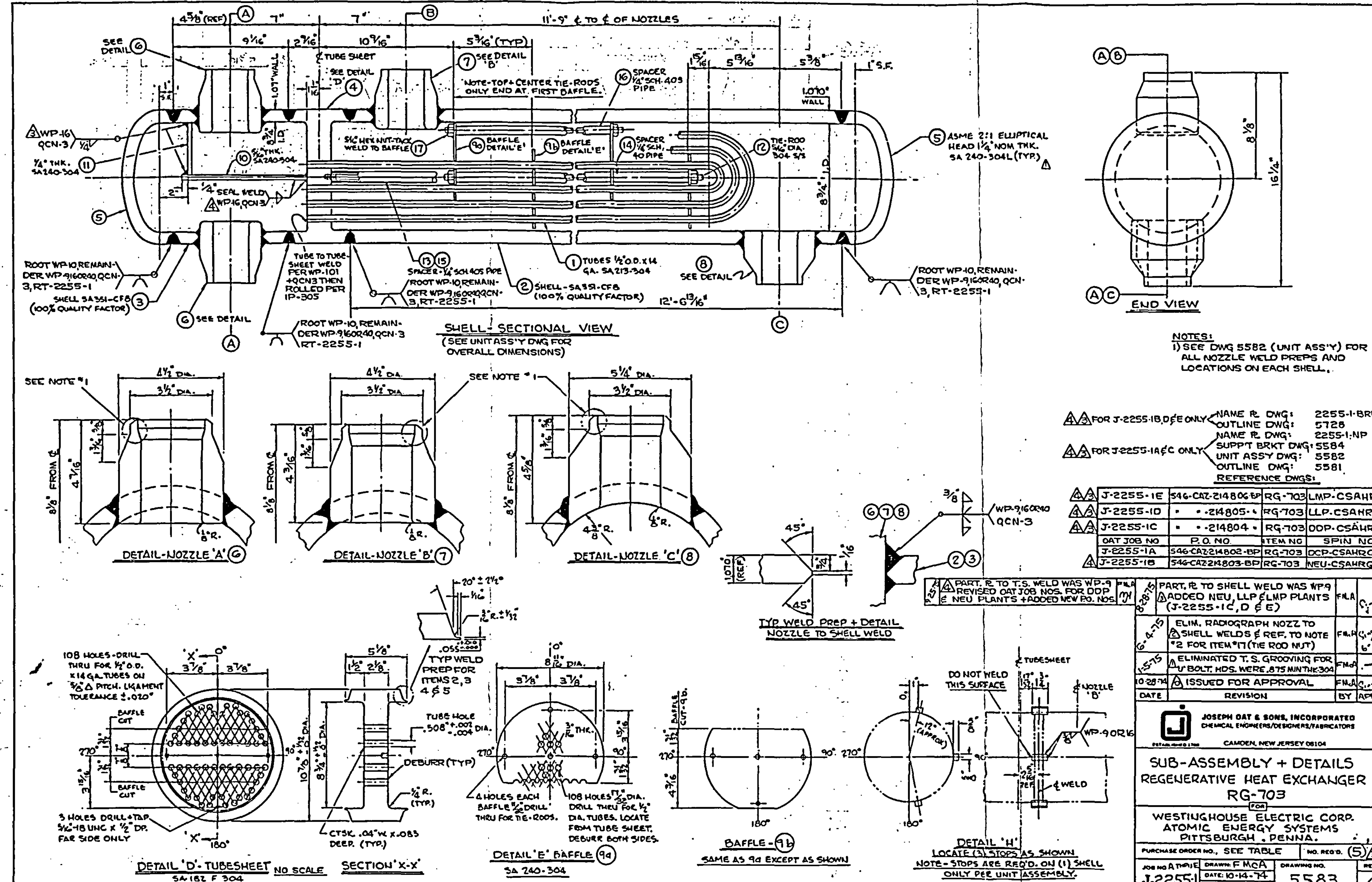
△3△2	J-2255-1E	546-CAZ-214806-BP	RG-703	LMP-CSAHRG
△3△2	J-2255-1D	546-CAZ-214805-BP	RG-703	LLP-CSAHRG
△3△2	J-2255-1C	546-CAZ-214804-BP	RG-703	ODP-CSAHRG
△3	J-2255-1B	546-CAZ-214803-BP	RG-703	NEU-CSAHRG
	J-2255-1A	546-CAZ-214802-BP	RG-703	DCP-CSAHRG
	OAT JOB NO.	P.O. NO.	ITEM NO	SPIN NO

JOSEPH OAT & SONS, INCORPORATED
CHEMICAL ENGINEERS/DESIGNERS/FABRICATORS
ESTABLISHED 1788
CAMDEN, NEW JERSEY 08104

REGENERATIVE HEAT EXCHANGER
10.90" O.D. x 14'-5 5/8" O.A.L.
UNIT ASS'Y RG-703
FOR
WESTINGHOUSE ELECTRIC CORP.
NUCLEAR ENERGY SYSTEMS
PITTSBURGH PA.

PURCHASE ORDER NO. SEE TABLE | NO. REQ'D. (5) △2
JOB NO. A THRU E: J-2255-1 | DRAWN: F McA | DATE 10-14-74 | SCALE: NONE | DRAWING NO. 5582 | REV. 4

DDE-00640960



NOTES:
1) SEE DWG 5582 (UNIT ASS'Y) FOR ALL NOZZLE WELD PREPS AND LOCATIONS ON EACH SHELL.

REFERENCE DWGS:

FOR J-2255-1B, D & E ONLY: NAME PL DWG: 2255-1-BRKT; OUTLINE DWG: 5728

FOR J-2255-1A & C ONLY: NAME PL DWG: 2255-1-NP; SUPP'T BRKT DWG: 5584; UNIT ASSY DWG: 5582; OUTLINE DWG: 5581

OAT JOB NO	P.O. NO.	ITEM NO	SPIN NO.
J-2255-1E	546-CAZ-214806-BP	RG-703	LMP-CSAHRG
J-2255-1D	" " -214805- "	RG-703	LLP-CSAHRG
J-2255-1C	" " -214804 "	RG-703	DDP-CSAHRG
J-2255-1A	546-CAZ-214802-BP	RG-703	DCP-CSAHRG
J-2255-1B	546-CAZ-214803-BP	RG-703	NEU-CSAHRG

DATE	REVISION	BY	APRD
9-25-75	(4) PART. PL TO T.S. WELD WAS WP-9 REVISED OAT JOB NOS. FOR DDP & NEU PLANTS + ADDED NEW P.O. NOS.	FMA	
8-28-75	PART. PL TO SHELL WELD WAS WP-9 (3) ADDED NEU, LLP & LMP PLANTS (J-2255-1C, D & E)	FMA	
6-4-75	ELIM. RADIOGRAPH NOZZ TO (2) SHELL WELDS & REF. TO NOTE #2 FOR ITEM #17 (TIE ROD NUT)	FMA	
1-5-75	(1) ELIMINATED T. S. GROOVING FOR U BOLT. HDS. WERE .875 MIN THK 304	FMcA	—
10-28-74	(0) ISSUED FOR APPROVAL	FMcA	

JOSEPH OAT & SONS, INCORPORATED
CHEMICAL ENGINEERS/DESIGNERS/FABRICATORS
CAMDEN, NEW JERSEY 08104

SUB-ASSEMBLY + DETAILS REGENERATIVE HEAT EXCHANGER RG-703

FOR

WESTINGHOUSE ELECTRIC CORP.
ATOMIC ENERGY SYSTEMS
PITTSBURGH, PENNA.

PURCHASE ORDER NO., SEE TABLE — NO. REQ'D. (5)

JOB NO A THRU E: J-2255-1 — DRAWN: F McA — DATE: 10-14-74 — SCALE: NONE — DRAWING NO. 5583 — REV. 4

DDE 00640961