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Report on Annealing of the Novovoronezh Unit 3 Reactor Vessel in the USSR

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

A U.S. delegation attended the thermal annealing operation of the Novovoronezh Unit 3 reactor vessel in the USSR to evaluate the Soviet reactor vessel annealing technology and to determine its applicability to PWR reactors in the U.S. Operations observed and described in this report include reactor vessel sample cutting, preparations for annealing, installation of annealing apparatus, and initial heatup of the reactor vessel. The annealing operation witnessed has been developed to a routine operation and appears applicable to U.S. PWRs. Key areas requiring further work to confirm applicability to U.S. reactors are discussed.

REPORT ON ANNEALING
OF THE
NOVOVORONEZH UNIT 3 REACTOR VESSEL IN THE USSR

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I. INTRODUCTION

This report is based on information obtained during the actual annealing of the reactor vessel in Novovoronezh Unit 3 in February and March 1991. The purpose of our visit was to evaluate the Soviet reactor vessel annealing technology and to determine its applicability to PWR reactors in the U.S for the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research.

There were two U.S. teams involved in witnessing the annealing at Novovoronezh Unit 3. The first team was in the USSR from February 2 through the 24, 1991, and witnessed the operation starting with the initial reactor vessel sample cutting, the setup and check-out of the annealing apparatus and its control system, the installation of the reactor vessel O.D. thermocouples and the associated monitoring station, the draining of the reactor vessel using the shielded cabin, installation of the annealing apparatus into the reactor vessel and the initial heat-up of the reactor vessel. The second team arrived after completion of the annealing and reflooding of the vessel since the cool-down occurred in about 2-1/2 days versus the normal 6-1/2 days. They reviewed the temperature traces during heat-up, annealing, and cool-down and witnessed the post-annealing sampling operation. They were in the USSR from March 2 through 9, 1991. The members of each team are listed in Appendix A.

While copies of detailed drawings were not provided, the Soviet personnel orally provided the technical details requested and allowed us to review the pertinent drawings and procedures. Accordingly, detailed dimensions and figures provided in this report should be taken as approximate. Also, the procedural steps described in this report are provided as an overall sequence and not as detailed step-by-step procedures. Further, this report does not attempt to cover the metallurgy, irradiation damage, etc., or the basic science behind reactor vessel annealings which has been previously supplied to the NRC by the Soviets. Basically, the report covers only the operational aspects of conducting an annealing of a reactor vessel in a pressurized water reactor plant. The Research Division of the NRC has indicated that the Soviet's annealing of their reactor vessels at 460°C (minimum) essentially removes 100% of the radiation damage and that re-embrittlement due to subsequent operation is reported to be no faster than initially experienced.

The signed protocol memoranda between the NRC and the Kurchatov Institute which summarizes the discussions and observations during these visit to Novovoronezh are covered in the NRC reports of these trips and are not duplicated herein.

II. ORGANIZATIONAL STRUCTURE

The Soviet organizational structure for development and implementation of reactor vessel annealings is based on the work of seven different Soviet agencies that were formed into one association called the "Annealing MOKHT." (MOKHT is an abbreviation for Interbranch Economic Association.) The agencies involved in the MOKHT for the reactor vessel annealing effort and their respective responsibilities are described below:

A. <u>OKB "Hydropress" (or "Gidropress")</u>

Hydropress is the main designer of the VVER nuclear systems and establishes the design criteria for the annealing apparatus. It is in effect the overall project manager for the annealing application. Its representative has overall charge of the annealing at the site and in essence serves as the Site Project Manager for the annealing effort.

B. <u>Kurchatov Institute of Atomic Energy</u>

The Kurchatov Institute performed the metallurgical research to establish the bases for the annealing (time, temperature) and the requirements for sampling, etc. In essence they provided the science behind the Soviet annealing process. The metallurgical samples removed from the Novovoronezh Unit 3 reactor vessel during the February/March 1991 annealing will be evaluated at the Kurchatov Institute.

C. <u>NPO "Cniitmash" (Central Institute of Heavy Construction)</u>

This is the designer, manufacturer and operator of the annealing apparatus and associated control equipment. Cniitmash personnel also record and process the data (temperatures associated with the annealing apparatus, power, etc.) from the apparatus' control system at the plant site.

D. VNIIAES - (All Union Research Institute for Nuclear Power Plant Operation)

VNIIAES performs the detailed heat transfer and stress analysis for the annealing operation from which the allowable heat-up and cool-down rates, as well as the allowable stress levels, were established. VNIIAES also mans the thermocouples monitoring station which records the temperature readings made on the outside of the reactor vessel during the actual annealing operation. Based on these readings they analyze and evaluate any unexpected temperature/stress levels during the actual annealing operation and, if necessary, would provide recommendations for adjustment in the heating parameters.

VNIIAES also developed the technique for hardness measurements and produced the equipment for this measurement.

E. TSNII KM "Prometei"

This is a metallurgical laboratory in Leningrad which does metallurgical sample evaluation. Its role in the Novovoronezh Unit 3 annealing operation was not discussed.

F. PO "Izhorskyi Zavod"

This is the reactor vessel manufacturer.

G. VVO "Atomenergoexport"

This is the official USSR export organization.

The above description of each organization's responsibilities and duties differs somewhat from that described in our November 22, 1991 report. The above is a more accurate description based on a better understanding of how the Soviet organizations actually function.

III. BACKGROUND ON NOVOVORONEZH UNITS 3, 4, AND 5

A. Novovoronezh Unit 3, which is a nominal 440 MWe PWR plant, was the first VVER-440 class nuclear plant. It started operating in 1971. Its sister plant, Novovoronezh Unit 4, started operating in 1972. Units 3 and 4 are built as an integral two-block unit (See Figure 1). During our visit to witness the annealing of Unit 3, Unit 4 was operating at full power within the same reactor building.

B. Novovoronezh Unit 3 was the first unit annealed by the Soviets. This first annealing was accomplished in 1987 at an actual temperature of 420°C (778°F) for 150 hours.

C. The first of the VVER-1000 class nuclear plants is Novovoronezh Unit 5. It is a 1000 MWe PWR type plant with a containment building similar to that used in the West; it started up in 1980. This class of plant also has redundant safety trains. At the beginning of our visit, Unit 5 was in the very final phase of a refueling outage and shortly thereafter returned to full power. Unit 5 is located on a separate site adjacent to Units 3 and 4.

IV. BACKGROUND ON ANNEALING OF SOVIET REACTOR VESSELS

A. Both the VVER-440 and VVER-1000 plants have reactor vessels made from ring forgings and as such they only have one circumferential weld in the active core zone. As a comparison, most U.S. reactor vessels are made from rolled plate and, as such, have vertical and circumferential welds in the active core zone. Most of the VVER-440 reactor vessels, including Novovoronezh Unit 3, are unclad. The VVER-1000 plants have clad reactor vessels.

B. Nine VVER-400 reactor vessels had been annealed prior to the annealing of the Novovoronezh Unit 3 reactor in February/March 1991. Seven of these vessels are unclad. Two weld-deposited clad VVER-440 reactor vessels in Eastern Europe have been annealed by the Soviets.

C. No VVER-1000 reactor vessels have been annealed to date. The Soviets indicated that they thought that these reactor vessels would only need annealing for life extension.

D. There have been three different time/temperature modes for reactor vessel annealing used by the Soviets:

Mode 1: (Used for the first annealing in 1987): Target Temperature of 430°C +/-10° (806°F +/-18°) for 150 hours.

Mode 2: Temperature of 475°C +/- 15° (887°F +/- 27°) for 150 hours.

Mode 3: Temperature of 475°C +/- 15° (887°F +/-27°) for 100 hours. (Used for the first time at Novovoronezh Unit 3 in February/March 1991).

E. The Soviets indicated that had they used Mode 2 or Mode 3 for the first annealing of Novovoronezh Unit 3 in 1987, they would not have needed to re-anneal Unit 3 again in 1991.

V. SITE PROJECT MANAGEMENT AND SCHEDULAR INFORMATION FOR THE ANNEALING OPERATION

A. The overall site project management set-up for the annealing operation at Novovoronezh, as near as we can determine, is shown in Figure 2.

B. The overall annealing sequence at Novovoronezh Unit 3 is shown on Figure 3. This figure also shows the critical path portion of the task for the annealing operations as well as support tasks.

C. For purposes of scheduling annealing operations in the U.S., where wall sampling and annealing operations would normally be done on a 24-hours a day, 7 days a week basis, the outage schedule shown on Figure 4 appears reasonable. Obviously, the amount and type of wall sampling can have a major impact on the schedule. [Note: The sampling and annealing on Novovoronezh Unit 3 were not critical path operations and as a result most of the operations were conducted on a one-shift-per-day basis, except for the actual annealing.]

D. The Soviets provided the following information on set-up of the annealing apparatus and its check-out:

Based on 3 shifts per day, they estimated it would take about 12 working days to:

1. Mechanically assemble the component and electrical parts of the annealing apparatus and its controls.
2. Install thermocouples on the apparatus, install movable insulating panels, calibrate meters and set up the test rig.
3. Test the annealing apparatus (i.e., heat-up) in the test rig.

VI. OVERALL SEQUENCE FOR ANNEALING OPERATION IN NOVOVORONEZH UNIT 3

A. Removal of Reactor Internals

The arrangement of the reactor cavity and the refueling canal in the VVER-440 reactor is such that the upper and lower internals cannot be stored in the refueling canal such as in U.S. PWRs. There is only a narrow canal (i.e., wide enough for one spent fuel assembly) between the reactor vessel refueling cavity and fuel storage area (see Figure 5). The VVER-440 reactor internals are removed as follows.

1. Upper Internals Removal

The upper internals package is removed from the reactor vessel in a cask which is also used to remove the lower core barrel assembly (see Figures 1 and 6). The cask is used to transport the upper internals to its special dry storage and servicing pit in the operating deck. Once the upper internals are in the pit, a shielded cap is installed with small ports so that all of the incore thermocouple stalks can protrude through the shielded cap for maintenance.

2. Lower Core Barrel Removal

The lower core barrel is removed and stored in a cask. The combined weight of the lower core barrel and the cask is 277 U.S. tons (251,280 kg). The rated capacity of the crane is 275 U.S. tons (250 metric tons). The measured radiation level about 6 inches from the cask surface and near the core center line was 36 mr/hr. (See Figures 1 and 6 for a depiction of this cask.) It should be noted that there is no valve or gate at the bottom of the cask; however, there are provisions for a bolted-on bottom cover and we believe that the bottom cover and bolting can support the internals weight. Also, supplemental temporary shield plates were provided around the lower end of the cask in the thinly shielded bottom section to reduce the radiation levels on the operating deck. The material of the lower core barrel section is an austenitic stainless steel (18%Cr, 12%Ni, 1%Ti .06%C).

B. Sampling Vessel Wall Prior to Annealing

1. Number of Samples

A total of six metallurgical samples were taken from the reactor vessel prior to annealing. Three base metal samples were taken near the core center line and at the high flux points on the vessel circumference. One of these samples is to be used to determine the original material properties (after being given a 1 to 2 hour anneal at 650°C). The second will be used to make impact specimens to determine the "as-found" irradiated impact properties. The third sample is to be held in reserve to help answer any questions that might arise during the evaluation.

Three other samples were taken from the reactor vessel circumferential seam weld in the active core zone region. This is the area of concern from a radiation damage viewpoint on these reactor vessels and the design of the annealing apparatus is based on only needing to anneal this weld area. These three weld samples serve the same purposes as the three samples of base metal described above.

NOTE: There are no vertical seam welds in the core zones of Soviet VVER reactor vessels as there are in most U.S. PWRs.

2. Sampling Machine

a. An EDM machine is used to cut the samples from the reactor vessel wall. The EDM machine requires use of a dielectric fluid bath for the electrodes (the fluid was described as being similar to kerosene). The cutting electrodes were copper and the operation was performed with the reactor vessel full of non-borated water (see Figure 7 for an overview of the EDM cutting machine and its cutting electrode arrangement). The dielectric fluid is retained around the cutting electrode by means of a box that seals against the wall of the reactor. When adequate sealing around the box is obtained, the water is pumped out of the box and replaced with dielectric fluid. This EDM process uses about 260 volts (DC) and approximately 10 amps. On occasions when the box seal did not properly seal, the dielectric fluid was released into the reactor vessel. The Soviets indicated that water in the reactor vessel is cleaned after annealing to remove any residue from such spillages and from the grinding operation involved with blending the sample locations in the vessel wall.

b. The cutting times required per sample were about 12 hours for the <u>base metal</u> samples and about 30 hours for the <u>weld sample</u> cuts. The overall cycle including: replacement of electrode, positioning, setup, check-out of the cutting rig, cutting the sample, removal of the rig, troubleshooting and repair; resulted in an overall average of about 1 sample per 1-1.5 days for the base metal samples and ~2 days for the weld metal samples. These cutting times were considerably longer than expected based on work previously done in the Novovoronezh Unit 2 vessel.

3. <u>Circumferential Location of Samples</u>
On VVER-440 reactors there are six local areas on the circumference where the neutron flux is the highest. Both base metal and weld metal samples were taken at these high flux areas. The three base metal samples were taken at an elevation near the core centerline and the three weld samples

were taken at the circumferential seam weld elevation in the active zone, all prior to the annealing operation. (See Figure 8.)

4. <u>Sample Locations on Circumferential Welds</u>
Since this vessel is <u>unclad</u>, the location of the circumferential weld can be seen with a TV camera and determined by the known vertical dimension from the reactor vessel flange. The approximate location of samples removed from the circumferential weld is shown on Figure 9.

5. <u>Weld Sample (or Template) -- Impact Specimen Size and Orientation</u>
The size of the circumferential seam weld sample removed from the reactor vessel by the EDM process is shown on Figure 10. Approximately 9 to 10 small size impact specimens will later be cut from each of the weld seam samples. The size of the weld impact specimens is 5 x 5 x 27.5 mm. This is the preferred specimen size for the Kurchatov Institute and is the normal size of weld seam samples. The thickness of the weld sample is slightly greater than for the base metal samples due to the larger volume of metal provided by the weld crown. The length dimension of the sample can vary, as shown on Figure 7 since the cut length is set by the second electrode used to free the sample. The thickness of approximately 5 mm was set by the reactor designer based on a maximum allowed wall penetration of 6mm.

6. <u>Base Metal Sample (or Template) -- Impact Specimen Size and Orientation</u>
The size of base metal sample as well as the size and orientation of impact specimens removed from the sample is shown on Figure 11. The base metal impact specimen size is slightly smaller than the weld metal specimen due to the curvature of the reactor vessel wall. As noted, the base

metal impact specimen size is 4 x 3 x 27 mm (standard size used by East Germany) and 12 specimens can be obtained from each sample.

7. Size of Sample Removal Cavity Left in the Reactor Vessel Wall
Figure 12 shows the size of the cavity in the reactor vessel wall as a result of removing the samples. After the post-annealing samples are removed, the sample cavities are blended by a remote-controlled grinder to minimize the stress concentration. (See Section of VI.F. below.)

C. Shielded Cabin - Hardness Testing and Draining of The Reactor Vessel

1. Each plant site has its own shielded cabin to allow people to work down inside the reactor vessel. (See Figure 13.) These cabins are of varying design depending on the needs of the individual plant sites. Normally to support the annealing operation, the shielded cabins are used for making hardness measurements in the active zone of the reactor vessel and for drying and cleaning the bottom of the reactor vessel. The cabin also serves as a shield to limit radiation from the reactor vessel when it is drained.

2. Also the shielded cabin at Novovoronezh Unit 3 has a positioning mechanism for the hardness tester. This mechanism was broken and the factory that builds replacement parts for this mechanism was shut down (or at least could not provide a replacement part soon enough). Since the Soviets were taking metallurgical samples from the vessel wall, they skipped the hardness test in the vessel. The Soviets indicated that they would be able to obtain the desired hardness measurements on the samples removed from the reactor vessel wall.

3. To remove the water from the reactor vessel, a sump pump (about 8" diameter and 1.5 ft long) with about a 3" diameter hose was first lowered to the bottom of the vessel. This sump pump discharged water out of the reactor vessel as the shielded cabin was lowered into the vessel and thus served as a shield as the vessel was emptied. This sump pump removed all but about a foot of water from the bottom of the vessel. At this point, the shielded cabin's top cover rested on ~1 ft high blocks on the reactor vessel flange; this allowed about a 10" radial annulus for access to remove the 3" hose and the first sump pump. After the first sump pump was removed, a smaller sump pump (~3" diameter and 9" long) with about 5/8" hose was lowered down the annulus, formed by the I.D. of the reactor vessel and the shielded cabin, to pump out most of the water remaining in the reactor vessel. After this small pump removed all the water possible, the door in the bottom of the shielded cabin was opened and a man working with rags and a bucket dried and cleaned the bottom of the vessel. When the shielded cabin was in this position, its body shielded the active zone of the reactor vessel and thus permitted access to the bottom of the vessel for about 15 minutes. (See Figure 13 for shielded cabin in reactor vessel and resulting radiation levels.)

4. The shielded cabin comes with various types of work sections and with a stack up of two different sections, depending on the tasks to be performed in the reactor vessel. Even the side panels of the shielded cabin can be changed to handle different tasks and toolings. Apparently these cabins have been used over the years for many types of inspections and repairs, (e.g., liquid penetrant testing of nozzles, repair welding of a lower core barrel guide, etc.). A complete shielded cabin assembly weighs in the order of 140 to 110 metric tons, depending on the number of shielded cabin sections and tooling being used.

D. Annealing Equipment and Operation

1. History of Annealing Apparatus Design and Use

a. Initially, an annealing apparatus was designed with an annealing zone of 3 meters. The "annealing zone" is the height of vessel where the wall temperature is held at a uniform annealing temperature. This design had seven layers of heaters stacked vertically as opposed to the three layers in current designs. When it was determined that only the one circumferential weld required annealing, an annealing apparatus with a 1-meter annealing zone was designed and built. The 3-meter annealing apparatus was never constructed.

b. The first annealing apparatus with a 1-meter annealing zone was used for the first five annealing operations.

c. A second annealing apparatus with a 1-meter annealing zone was built because it appeared that more than one plant would need to be annealed simultaneously. This second unit's design incorporated lessons learned from experience with the first assembly. The second unit is simpler to use (fewer moving parts, a simpler mechanism for deployment of thermocouples, improved maintenance arrangement for heaters, etc.). This second annealing apparatus has been used for four annealings prior to its use for the current annealing at Novovoronezh Unit 3 in February/March 1991.

2. <u>Mechanical Description of Annealing Apparatus</u>

The annealing equipment consists of the following major components:

a. annealing apparatus

b. electrical transformers, and

c. control cabinets.

These components are described below.

a. <u>Annealing Apparatus</u>:

The annealing apparatus which has a total weight of 28.7 tons (26 metric tons) consists of the following subcomponents and features (see General Arrangement, on Figure 14).

- top shield plate
- center lifting column
- lift shaft
- heater panels
- retractable thermocouples
- insulation panels
- transportation of annealing apparatus
- mounting stand

Each of these is described below:

(1) The <u>top shield plate</u> weighs 17.6 tons (16 metric tons). The functions of the plate are to support the full weight of the annealing apparatus on the reactor vessel flange, to center the heater in the vessel, and to radiologically shield workers from the dry reactor vessel. There are six centering blocks which are bolted to the reactor vessel flange (using the existing tapped holes in the flange) to provide guiding for the heater when it is seated on the vessel flange.

(2) The <u>center lifting column</u> is a stainless steel pipe that connects the heater assembly with the top shield plate. This pipe houses the movable lift shaft and provides structural attachment points for supports for the electrical bus bars, and for framing that extends outward to the radius of the ring in the reactor vessel that separates the inlet and outlet nozzle elevations.

(3) The movable lift shaft has a lifting eye welded to its upper end and runs through the center lifting column. The functions of the lift shaft are to support the annealing apparatus whenever it is lifted by a crane, and to retract and extend the thermocouples and insulation panels when the annealing apparatus is lifted and placed in position, respectively.

(4) There are 54 heater panels, each about 24 inches high by about 18 inches wide. The panels are grouped into three elevations (upper, middle, and lower) and three 120° sectors, so that there are nine individually controllable groups with six heater panels in each group. Each of the nine groups is supplied with electrical power from a transformer (nine transformers total). The heater panels are fixed in position about eight inches (200 mm) away from the reactor vessel surface. The panels consist of a backing plate, an insulation layer, and a series of heater bars facing the reactor vessel surface. Panels are easily replaceable by threaded fasteners.

The designers of the annealing apparatus (Cniitmash) said that the heating unit is designed to operate at up to 1,200 kW, but that the maximum power level ever used is about 750 kW [Note: The power ratings of the transformers powering the heater would limit the power to about 800 kW]. During heat-up for annealing, about 225 kW is actually used and only about 150 kW is used while holding the vessel at the steady state annealing temperature.

(5) There are nine thermocouple mechanisms, one at the center of each of the nine heater groups. The function of the mechanisms is to press thermocouples against the inside surface of the reactor vessel in order to measure the vessel temperature during heat up, annealing, and cool down. At each mechanism, two metal rods, about 3/8 inches in diameter, and about 12 inches long, with a 1/2 inch (approximate) spherical ball end, are clamped into the mechanism. The thermocouple is encased in the spherical ball end. The mechanism uses a weight and lever arrangement to rotate the thermocouple rod outward from the heater assembly such that the ball end rests against the vessel. The thermocouple rods are rotated up to their retracted position when the movable lift shaft is pulled upward by the lifting eye, and they

rotate to their extended positions by the force of the individual weights (one at each mechanism) when the lifting eye is released and the lift shaft moves downward. Each of the nine mechanisms can rotate outward independently to ensure contact is made with the vessel surface. Within a given mechanism, the two thermocouple rods move together, since they are clamped to a common block.

(6) The thermal insulation panels are located just above the upper elevation of heater panels. Their function is to restrict the flow of heated air up along the annular gap between the heaters and vessel. The insulation panels are segmented radially to permit them to move during installation and removal of the heating rig. They are pulled in toward the center of the heating rig to provide clearance during installation and removal, and are rotated down into place against the vessel wall during vessel annealing. Like the thermocouple rods, the motion of the insulation flaps is driven by the movable lift shaft.

(7) The transportation of the annealing apparatus between plant sites is by truck or rail. The heater assemblies are unbolted as 120° segments, which are each laid onto shipping cradles. The center lifting column and liftshaft are shipped as one integral piece. The top shield plate is separated into three slabs for transportation.

(8) The mounting stand is transported by truck to the plant and assembled in the reactor hall. The annealing apparatus is assembled on the stand and tested prior to installation in the reactor vessel. The stand provides a lower, middle, and upper deck from which workers can access the annealing apparatus. Segments of the middle and upper decks are removed to allow the annealing apparatus to be lifted and translated horizontally away from the mounting stand. While in the mounting stand, the annealing apparatus is supported at the top shield plate.

b. Electrical Transformers:

A single 1000 kVA transformer (cabinet size approximately 4′ x 6′ x 6′) was provided to step down the plant's 6,000 VAC, 50 HZ power to 380 VAC for use by the annealing equipment. The 380 VAC power is then stepped down to less than 100 VAC by nine 90 kVA transformers powering the nine groups of heater panels. The cabinet size for these transformers is about 2′ x 3′ x 4′.

c. Control Cabinets:

Six cabinets, each about 30" wide, 24" deep, and 72" tall, house the strip chart recorders, gages and controllers for monitoring and controlling the annealing apparatus.

3. Electrical and Control Aspects of Annealing Apparatus

a. The electrical and control systems are shown schematically in Figure 15. The Soviets take 6,000 VAC power from a source that is tied into the power station's emergency diesel generators. A reliable power source is important since some power is needed to prevent the 30°C/hour (54°F/hour) cool-down limit from being exceeded for the portion of the cool-down between 475°C (887°F) and 300°C (572°F). For example, during an annealing operation at an Armenian plant, power was lost momentarily during a lightning storm, and the station's diesel generators provided back-up power.

b. There are two thermocouples in each 120° sector at each elevation. One thermocouple is used for temperature control and the other is a back-up. The thermocouple signal is plotted on a strip chart recorder and is input to a microprocessor based controller, which in turn drives thyristors controlling the voltage to the primary side of the transformers. The transformers nominally provide 68 VAC to the middle bank of heaters and 91 VAC to the upper and lower banks.

c. There are a total of four redundant levels of control. The microprocessor-based controller has a complete backup (i.e., a redundant microprocessor-based controller). If the microprocessor-based controls fail, a hard-wired electronic controller is used to control the thyristor, based on comparison of the thermocouple signal and a manually-selected temperature set point. If the thyristors fail they are bypassed, and the heaters are switched on and off with relays at the primary side of the transformers.

d. The microprocessor based controller is a general purpose device (i.e., it was not custom built for the annealing application) that is programmed to ramp the vessel temperature up and down during the annealing operation. The program does not include any interlocks or automatic steps to stop the heat-up or cool-down based, for example, on temperature distributions or heater amperages.

e. There are nine additional thermocouples welded into the heating elements (one per heater group). These temperatures, which are indicated along with amperages drawn by the heater groups, are used by the operators to confirm proper operation of the heaters, but are not process control parameters.

4. <u>Installation of Thermocouples on the Outside of the Reactor Vessel for Monitoring the Annealing Operation</u>

a. VNIIAES performs the heat transfer and stress analysis for the annealing operations and selects the desired location of the thermocouples (in conjunction with the plant staff) on the outside surface of the reactor vessel. The plant staff provides the devices to mechanically spring-load the thermocouples against the reactor vessel walls.

b. The locations where these thermocouples are installed are shown on Figure 16. Except for the two lowest locations (positions 5 and 6), three thermocouples are installed at each elevation, 120 degrees apart. At each of the lowest two locations, only one thermocouple is installed.

c. There are a number of thermocouples permanently welded to the reactor vessel. These thermocouples read out only in the control room and are only used as a back-up when problems or questions arise with the special thermocouples installed for the annealing.

d. The plant staff provides the recording and monitoring set-up for thermocouples on the outside surface of the reactor vessel. VNIIAES mans this monitoring operation during the annealing and has the ability to perform real-time on-site stress analyses using the measured temperature readings, if they deviate from previously estimated limits or expected conditions. (i.e., they can take actual measured temperatures and predict the stress levels to determine if deviant conditions are within acceptable stress limits).

5. <u>Checkout and Testing of the Annealing Apparatus</u>

Following assembly of the annealing apparatus in the reactor hall, detailed inspections and tests were performed prior to installing the apparatus in the reactor vessel. The site annealing project manager indicated that the time required for setup of the checkout stand, assembly, inspection and testing of annealing apparatus can vary, but that they had allowed about 12 work days (24 hours per day) for this checkout work.

Preventive maintenance was done on the control cabinets prior to testing. The amount of maintenance done was greater than usual due to concerns that the equipment might have been damaged by the weather. [<u>Note</u>: This equipment was shipped by train, uncrated, off-loaded, and left outside exposed for a week or two before being moved inside the power plant].

After the preventive maintenance, electrical continuity checks of major components and cabling were performed. Insulation pads were then hung on the annealing apparatus about 10 inches away from the heater panels. The insulation pads are used to avoid heating the reactor hall excessively. One insulation pad was left off to allow the heaters to be seen during heat-up tests. Power was then applied to a number of heater groups simultaneously. The thermocouples embedded in the heating panels and the thermocouples which contact the reactor vessel were monitored along with amperage drawn by the heaters. A final test was then performed in which individual heater groups were powered up and down and the thermocouple signals were monitored to confirm that the thermocouples were wired correctly.

The temporary insulation pads were then removed and the electrical leads to the annealing apparatus were disconnected and coiled on the top shield plate in preparation for the installation of the apparatus into the reactor vessel.

6. <u>Requirements for Isolating the Reactor Vessel to Prevent Water Contacting the Vessel During Annealing</u>

a. Unlike the U.S. PWRs, the annulus between the reactor vessel flange and the refueling cavity liner in Soviet plants is permanently sealed by a stainless steel welded bellows; thus refueling cavity water cannot drain down on the outside surface of the reactor vessel.

b. The various procedural steps for preparing the reactor vessel for annealing and for preventing water from reaching the hot reactor vessel wall are provided as Appendix B to this report.

7. <u>Installation of the Annealing Apparatus into Reactor Vessel and Associated Radiation Levels</u>

a. After the reactor vessel has been dried, the shielded cabin is removed, leaving an empty and dry reactor vessel (see discussion in Section V.C. for draining of vessel). The radiation levels with the reactor vessel dry and empty and ready for receiving the annealing apparatus are shown on Figure 17.

b. The sequence of rigging to install the apparatus was as follows:

- The auxiliary hooks on the 277 U.S. ton crane were used to remove two rail sections on the side of the mounting stand used to check out the apparatus.
- The 277 U.S. ton (250 metric tons) crane was used to remove the shielded cabin from the reactor vessel and store it in a pit in the operating deck.
- The 277 U.S. ton crane was then employed to lift the annealing apparatus and move it to the reactor cavity. (The centering blocks had been installed on the reactor vessel flange previously).

These rigging moves went smoothly and quickly.

c. Radiation levels were monitored during these moves and personnel were kept clear of the reactor cavity area until surveys were done. As indicated in Figure 13, the maximum radiation level at the top of the refueling cavity was 1.6 r/hr. Just back from the face of the refueling cavity, the radiation level was 80 mr/hr. The crane operator used the operating station at the opposite end of the reactor hall while removing the shielded cabin. Radiation levels near the crane operating station nearest the Unit 3 reactor vessel were about 4 mr/hr. The crane operator then moved to the operating station nearest the vessel to move the annealing apparatus into the reactor vessel.

d. While the crane operator initially lowered the annealing apparatus into the reactor cavity, no personnel were standing at the edge of the cavity. It appeared the crane operator used index marks visible on the crane's support beam to approximately locate the crane's hook (and the annealing apparatus) at the reactor vessel's centerline. Once the annealing apparatus was lowered such that the heater assembly was just above the reactor vessel flange, personnel moved to the edge of the reactor cavity and steadied the annealing rig by hand to eliminate its slight swinging motion. These personnel then provided hand signals to the crane operator to fine-tune the position of the apparatus as it was lowered past the reactor vessel flange.

8. <u>Initial Heat-up of the Reactor Vessel</u>

a. After the annealing apparatus is installed in the reactor vessel and the thermocouples (TCs) and the insulation panels are moved out to contact the I.D. of the reactor vessel, the heat-up is initiated. The heat-up rate as measured by all 18 TCs on the annealing apparatus that contact the ID of the reactor vessel in the zone to be annealed are limited to 20°C (36°F) per hour.

b. All 18 of these same TCs on the I.D. of the reactor vessel must be within 50°C (90°F) of each other (the difference between the highest and lowest of the temperature readings must be less than 50°C). Normally, there is less than 20°C (36°F) difference between these TCs. These 18 TCs are input to the annealing apparatus control panels for control and recording purposes. The annealing apparatus controls are operated and monitored by Cniitmash personnel throughout the annealing operation.

c. VNIIAES personnel attend the recording station for the thermocouples on the outside of the reactor vessel. This thermocouple recording station is separate and is not a part of the control system of the annealing apparatus provided and monitored by Cniitmash. As the reactor vessel is heated or cooled, VNIIAES personnel make real-time evaluations of the actual O.D. thermocouple readings versus the predicted temperatures. The O.D. thermocouples at any one elevation must be within 50°C (90°F). If thermocouple readings vary from the predicted or deviate from the 50°C limit for one elevation, VNIIAES personnel use a personal computer at the site to make real-time analyses to confirm that the specific deviation from expected temperature patterns are within acceptable stress limits.

d. During the initial heat-up of Novovoronezh Unit 3, water was found in a hot leg pipe between the reactor vessel and the loop isolation valve. The heat-up was stopped and the reactor vessel held at 100°C. The source of the water was found to be the pressurizer spray line which had not been properly vented. At this point in the annealing operation, the plant management required the operating staff go through all isolation, venting and draining procedures again to ensure that all water was removed and isolated from the reactor vessel before proceeding with the heat-up.

9. Steady State Annealing Conditions

The one meter annealing zone was held at 475°C +/- 15° (887°F +/-27°) for 100 hours. Once steady state conditions are obtained, only a minimum amount of power is needed to the center section of heaters; the top and bottom heater sections do most of the heating during this condition. See Figures 18 and 19 for profiles of vessel, shield tank, and concrete temperatures as measured during the annealing of Novovoronezh Unit 3.

10. Cool-down Following the Annealing and Subsequent Reflooding of the Vessel

a. While cooling down between 475°C (887°F) and 300°C (572°F) some limited power input [illegible] provided to the annealing apparatus to keep within the 30°C/hr (54°F/hr) per hour cooling down limit. In response to the questions regarding a heater section failure during this portion of the cool-down cycle, the Soviet engineers indicated that only limited power is needed from any of the other two heater sections in the same quadrant of the apparatus to provide sufficient power to keep the cool-down within limits.

b. At 300°C (572°F) as measured by the reactor vessel I.D. thermocouples, the power to the annealing apparatus is turned off and the apparatus is slightly raised to help cool the vessel down. This raising of the annealing apparatus automatically retracts both the thermocouples and the moveable insulation panels from contact with the inside surface of the reactor vessel wall. At this point, control of the cool-down is switched to thermocouples on the outside surface of the reactor vessel which are recorded by the VNIIAES recorders.

c. When the vessel wall temperature reaches 70°C, (158°F) water is re-introduced to the reactor vessel. Just before the water is re-introduced, the annealing apparatus is finally lifted completely out of the reactor vessel cavity. During the later stages of the cool-down, the raised apparatus serves as a shield to minimize the radiation level prior to refilling the vessel with water. In regard to their cool-down between 300°C and 70°C, the Soviets have indicated that this takes much longer than desired or necessary and they are trying to find ways to help speed up this portion of the operation.

d. It required approximately 63 hours to actually cool the Novovoronezh Unit 3 reactor vessel from 475°C to ~70°C when water was re-introduced into the vessel. The temperature profile 32 hours into the cool-down phase is shown on Figure 18.

E. Sampling Vessel Wall After Annealing

After annealing, two weld metal samples were taken from one of the high fluence locations as described in Section VI.B.3 of this report. These metallurgical samples are to be the same size as the "before" samples described in Section VI.B.5 (Figure 10).

F. Blending of Reactor Vessel Wall Sample Cavities

The shielded cabin is equipped with a grinding tool which is used to blend the metallurgical sample cavities made in the wall of the reactor vessel. Figure 20 shows the dimensional requirements for blending the sample cavities. These requirements were established by Hydropress. This blending operation takes place only after all vessel sampling work is completed.

VII. APPLICATION OF SOVIET ANNEALING TECHNOLOGY TO U.S. REACTOR VESSELS

A. Summary of Areas Requiring Changes, Investigations, or Modifications to Apply Soviet Annealing Techniques to U.S. Reactor Vessels

The following is a summary of the four key areas requiring investigations and/or modifications to apply the Soviet annealing technology to U.S. PWRs.

1. Annealing Apparatus

   The existing Soviet annealing apparatus, due to its size, is not directly usable in the U.S. plants. Accordingly, both the diameter and height of the apparatus' annealing zone, will need to be changed to accommodate U.S. reactor vessels with both circumferential and vertical seam welds. However, rescaling of the Soviet annealing apparatus appears to be a straightforward task.

2. Stress Analysis to Support the Annealing Operation

   The heat transfer and stress analysis for support of an annealing operation will be significantly more involved in the U.S. plants than in Soviet plants since the annealing zone will cover a greater height, possibly approaching 12 feet in most U.S. plants. This longer annealing zone is due to the vertical weld seams in U.S. vessels. The Soviets only need to anneal one circumferential seam weld (i.e., Soviet vessels do not have vertical seam welds). As a result of the longer annealing zones, U.S. plants will most likely experience much steeper thermal gradients (thus high stress levels) through the vessel nozzle areas and the core barrel guide lug regions of the vessel wall. If resulting stresses are too high, it may be necessary to cut the reactor coolant pipe, free up the bottom incore instrument piping, etc., to accommodate the thermal movement and thus reduce the thermal stresses. Another alternative may be to reduce the annealing zone height and therefore not fully anneal the entire active zone. [Note: This might require more frequent annealing operations.] It is our opinion that the first effort which should be undertaken to determine how to apply the Soviet annealing technology to U.S. plants is to perform a stress analysis using the longer annealing zone required in U.S. plants. Such analysis will probably be required for each type of U.S. reactor. Once such an analysis has been conducted, then evaluations can be made as to what actions must be taken to accommodate the resulting stress levels. In making such a scoping analysis we would suggest using the Soviet annealing temperature used for Novovoronezh Unit 3 in 1991, (i.e., ~887°F) The final annealing temperature for U.S. vessels may be somewhat different, but this should be a reasonable starting point.

[Note: As an indication of the increased stress levels that can be encountered with a longer annealing zone, see Figure 21 which shows that the stress levels increased by over a factor 2 when going from a 1-meter to a 3-meter annealing zone in a VVER-440 Reactor Vessel.]

3. Draining of Reactor Vessel with Reactor Internals Removed
   a. A system must be developed to allow the reactor internals in U.S. PWRs to be removed and stored and at the same time permit draining of the reactor vessel so it can be annealed (see Section VII.B below for further discussion of this issue).

   b. Higher radiation levels expected with the clad U.S. reactor vessels may require some different sequences and shielding devices during draining than those used by the Soviets. Based on the increased nickel content in the clad, the radiation levels could be up to ten times higher in U.S. plants than for an unclad Soviet vessel as shown on Figure 17. We believe that these projected higher radiation levels will need to be addressed but will not have major impact on applying the Soviet annealing technology to U S. plants.

4. <u>Monitoring of Thermal Stresses during the Annealing Operation in U.S. plants</u>

   a. Schemes for installing the thermocouples (TCs) on the outside of the U.S. reactor vessel to monitor the thermal gradient (i.e., to allow calculating thermal stresses) during the annealing operation will need to be developed. Such TC installations will most likely be unique to each U.S. plant because the reactor vessel insulation, the reactor vessel cavity arrangement, and physical access are very different from plant to plant. While such thermocouples can be installed in U.S. plants, we believe it will require more of an engineering effort due to:

      - The higher thermal stress levels that will most likely be encountered in annealing U.S. plants may require more exact locations of TC's.

      - More limited physical access to areas of interest in U.S. reactor vessel cavities for locating TCs to monitor points of highest stress level during annealing operation.

      - Higher radiat[illegible] and contamination levels in areas where TCs need [illegible] be installed on the outside of the reactor vessel. [<u>Note</u>: Leakage of U.S. refueling cavity seals may have contaminated areas around the reactor vessel and its insulation where TCs need to be installed; Soviet plants, on the other hand, have welded bellows for the refueling cavity seal and thus have not experienced such leakage and contamination around the reactor vessel and its insulation where they install TCs.]

b. Since the thermal stress levels during heat-up, steady state annealing, and cool-down of U.S. reactor vessels will most likely be closer to allowable stress limits than experienced in the Soviet annealing (i.e., principally due to longer annealing zones in U.S. vessels), thermal stress monitoring operations and the capability to perform stress analysis during actual on-site annealing operations will probably need to be more extensive than for the Soviet plants.

While the above four areas of concern will require more effort in the U.S. plants, they should not be considered "show stoppers" in applying the Soviet annealing technology. The reasons the Soviet plants have not had to face these problems to the same degree are as follows:

- Their vessels do not have vertical weld seams and thus require a short annealing zone.
- Soviet reactor vessels have about 9" to 12" further distance between the top of the active core and the nozzles than do U.S. vessels, thus providing more distance between the annealing zone and the nozzles.
- The Soviet plants annealed to date are basically one standardized design, while each U.S. plant tends to be unique.
- The reactor vessel in Soviet plants can be readily drained when the reactor internals are removed, whereas this is not true for most U.S. PWRs.
- The outside surface of the Soviet reactor vessel has fairly good access for installing TCs to monitor the

annealing operation, and it has minimum contamination and radiation levels in the reactor vessel cavity where the TCs need to be installed.

B. Possible Concepts to Allow Draining of Reactor Vessel in U.S. PWR's

1. Storage of Reactor Internals during Annealing

Both the Soviet VVER-440 and the VVER-1000 PWRs have features which allow the reactor vessels to be readily drained when the reactor internals are removed and stored. These features greatly facilitate annealing the Soviet reactor vessels. In the VVER-440 the internals are handled by a combination of a shielded cask and a separate dry storage pit. (See Section VI.A and Figures 1, 5 and 6). The VVER-1000 refueling cavity is similar to the U.S. PWRs cavity but these Soviet plants have a gate which allows separation of the reactor internal storage area and the reactor vessel (See Figure 5).

In U.S. PWRs, when the reactor internals are removed and stored in the deep end of the refueling cavity, the cavity normally cannot be drained in the area immediately around the reactor vessel. To allow draining of the reactor vessel, a large diameter cofferdam (14′ to 16′ diameter and about 25′ to 27′ long) could be made which bolts and seals to where the present refueling plate seals on the refueling cavity liner. Once the cofferdam is in place, it will retain water for shielding in the portion of the refueling cavity where the reactor internals are stored and still allow the reactor vessel to be drained.

[Note: This will require some special design features to allow the refueling seal plate to remain in place until the cofferdam is in place and sealed. After the cofferdam is in place, the refueling seal plate could be removed. By sealing the cofferdam to the refueling cavity liner, concerns with seal plate leakage during annealing can be eliminated. Further, the movement and heating of the reactor vessel flange during annealing does not affect the seal of the cofferdam. Current U.S. refueling seal plates frequently leak a limited amount; this is not a problem for refueling but could result in problems during annealing. The Soviets make a major effort to ensure that cold water does not get on the hot vessel during annealing].

Another possible alternative scheme to allow drainage of the reactor vessel is to use two shielded casks or shield blocks in the refueling cavity for temporary storing and shielding the internals. Such a concept will be dependent on the polar crane capacity and laydown space in containment or, in the case of shielded blocks, the cavity arrangement. For the larger size PWRs, it is estimated that the cask and core barrel would weigh in the order of 300 to 400 tons plus supplemental shielding (the Soviet cask and lower internals for an 8 ft. core zone of a VVER-440 reactor weighed 277 U.S. tons -- see Figure 6).

2. <u>Radiation Levels when Draining Water from U.S. Reactor Vessels</u>

If radiation levels are too high to allow complete draining of the U.S. reactor vessels, one of the two following sequences can be considered:

a. A shielded deck plate can be built and suspended from the refueling canal walls to cover the top of the cofferdam. A sump pump and hose could be lowered through the shielded plate to remove the bulk of the water in the vessel. Long handled tools could then be used through a small port in the shielded deck plate for final drying of the vessel. The shielded deck plate could be removed just before the annealing apparatus is installed in the vessel.

b. Another draining approach is to provide a sump pump just beneath the annealing apparatus. The reactor vessel could be pumped dry as the annealing apparatus is lowered. This is similar to the scheme the Soviets use when installing the shielded inspection cabin. In such a conceptual arrangement, the shielding provided by the annealing apparatus should reduce the radiation levels during its installation. Specifically, its top cover section (see Figure 14) should minimize any concern about radiation levels when the apparatus is completely installed in the reactor vessel. Such an approach would require some additional modifications to the Soviet annealing apparatus, (e.g., the inclusion of special ports in the top shield cover to allow access to the bottom of the vessel for final drying of the vessel and removal of pumps and hoses before the heaters are turned on).

C. Sizes of Typical U.S. PWR Reactor Vessels

For information on the sizes of typical U.S. PWR reactor vessels see Figure 22. For information on the Yankee Rowe reactor, which the NRC indicates may be an early candidate for annealing, see Figure 23. For comparison with the Soviet VVER-440 reactor vessels see Figure 16. With regard to comparing these vessels, remember that the U.S. vessels have both a circumferential and vertical seam welds which will most likely require an annealing zone covering most of the active core height. The Soviets only need to anneal one circumferential weld seam in their vessels and as a result they have a much shorter annealing zone.

VIII. SUMMARY AND CONCLUSIONS

A. Summary

1. During February and March 1991, two U.S. teams witnessed the successful annealing of the reactor vessel in the Novovoronezh Unit 3, which is a VVER-440 class pressurized water reactor plant. The Soviets were open and helpful in assisting the U.S. team in witnessing and understanding what was actually involved in conducting an annealing operation in their power plant. Further the U.S. personnel were given unrestricted access for photographing and video taping the annealing operation.

2. The following briefly summarizes the differences between Soviet and U.S. PWR's from the standpoint of an annealing operation. Each of these makes annealing of U.S. PWRs somewhat different.

   - The Soviet annealing apparatus has a shorter annealing zone since their reactor vessels only have a circumferential weld in the core zone which requires annealing; whereas our reactor vessels have both circumferential and vertical welds and thus require a much longer annealing zone.

   - The Soviet plants have lower stress levels during annealing due to the shorter annealing zone. The short annealing zone avoids effecting the vessel nozzle and core barrel guide block regions. In the U.S. plants these regions will be effected by the longer annealing zone and as a result will experience higher stress levels during annealings.

   - The Soviet plants have features which allow reactor vessels to be readily drained after the reactor internals are removed. In U.S. PWR's, the water cannot normally be drained from the reactor vessel when the internals are removed, since the water is required to shield the internals which are stored in the refueling canal.

   - In Soviet plants, cavities surrounding the reactor vessel are reasonably accessible and have low radiation and contamination levels in the areas where special thermocouples are installed to allow evaluation of annealing induced stresses. In most U.S. plants, these cavities are less accessible and more contaminated due to leaking refueling cavity seals.

- The Soviet plants generally have lower radiation levels on key components and lower radiation levels from a dry reactor vessel since their vessels are unclad.

- The Soviet plants being annealed are a standardized plant design which results in a common time-temperature requirement for annealing, one basic annealing apparatus, one basic stress analysis, one basic stress monitoring technique, and same basic procedures for all plants. Most of these areas will probably have to be custom tailored for each U.S. plant.

B. Conclusions

1. The Soviet annealing equipment is rugged, simple, easily maintainable and appears quite reliable. Its control system is straightforward and has considerable redundancy.

2. The annealing operation used by the Soviets on the VVER-440 reactor vessels has been basically developed to the point where it is a routine maintenance operation.

3. The basic Soviet annealing technology appears applicable to U.S. PWR's.

4. The following is a summary of the key areas requiring investigation and/or modification to confirm the applicable of Soviet annealing technology to U.S. PWR's.

   - Perform a stress analysis on U.S. vessels to determine the impact of a longer annealing zone which is needed to cover the vertical seam welds in U.S. vessels.

   - Develop techniques to allow draining of U.S. reactor vessels when reactor internals have been removed and to deal with the expected higher radiation levels from a clad reactor vessel.

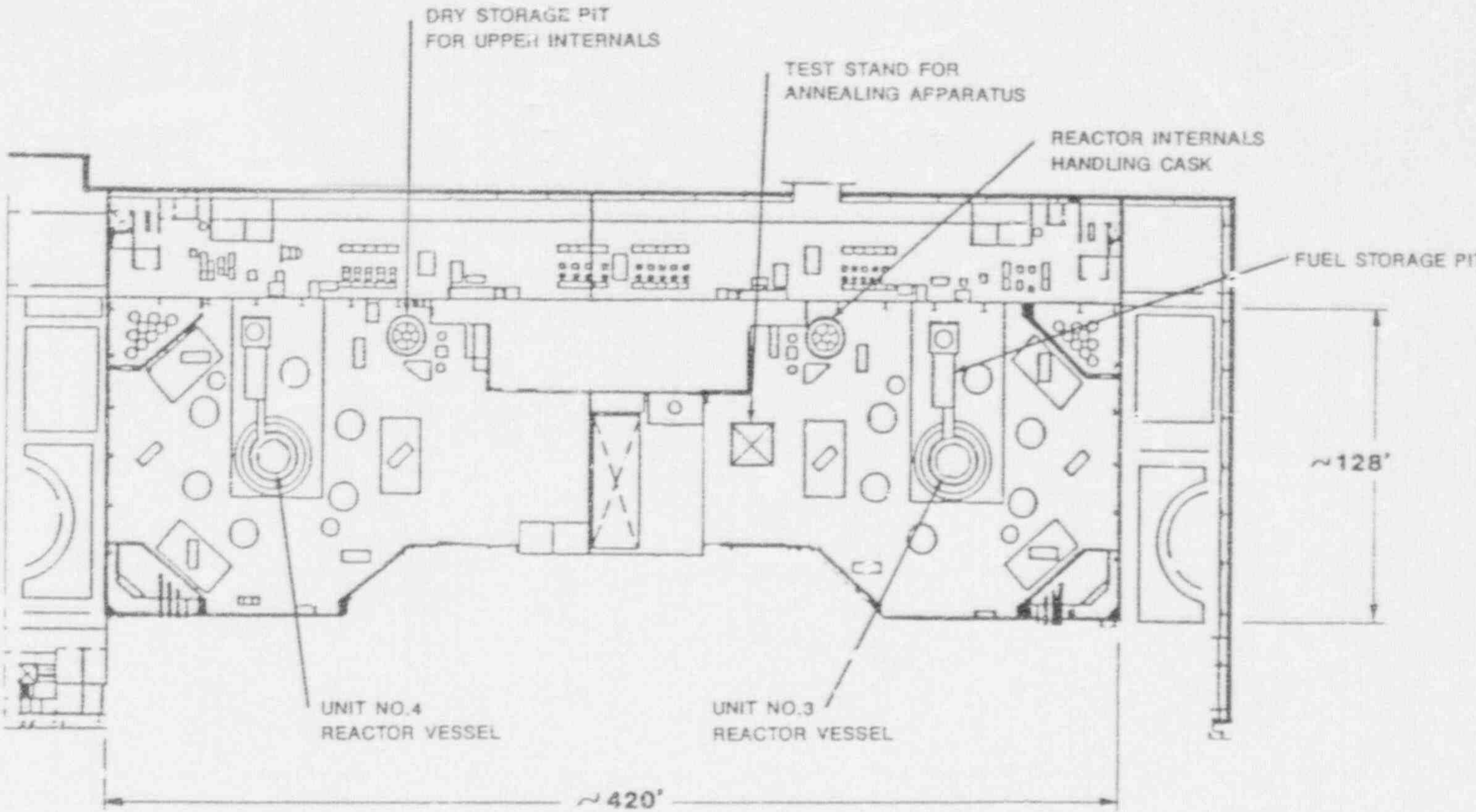
   - Develop a practical means of monitoring of thermal stresses during annealing; this will be more involved and extensive on U.S. reactor vessels since annealing in our plants will require operating closer to the allowable stress limits.

- Develop a design of an annealing apparatus for U.S. reactor vessels which have different diameters and annealing zone lengths than the Soviet vessels.

While the above four areas will require somewhat different approaches and/or techniques to resolve, in our judgement they should not be "show stoppers" in applying the basic Soviet annealing technology to U.S. PWR's. These issues are discussed in further detail in Section VII of this report.

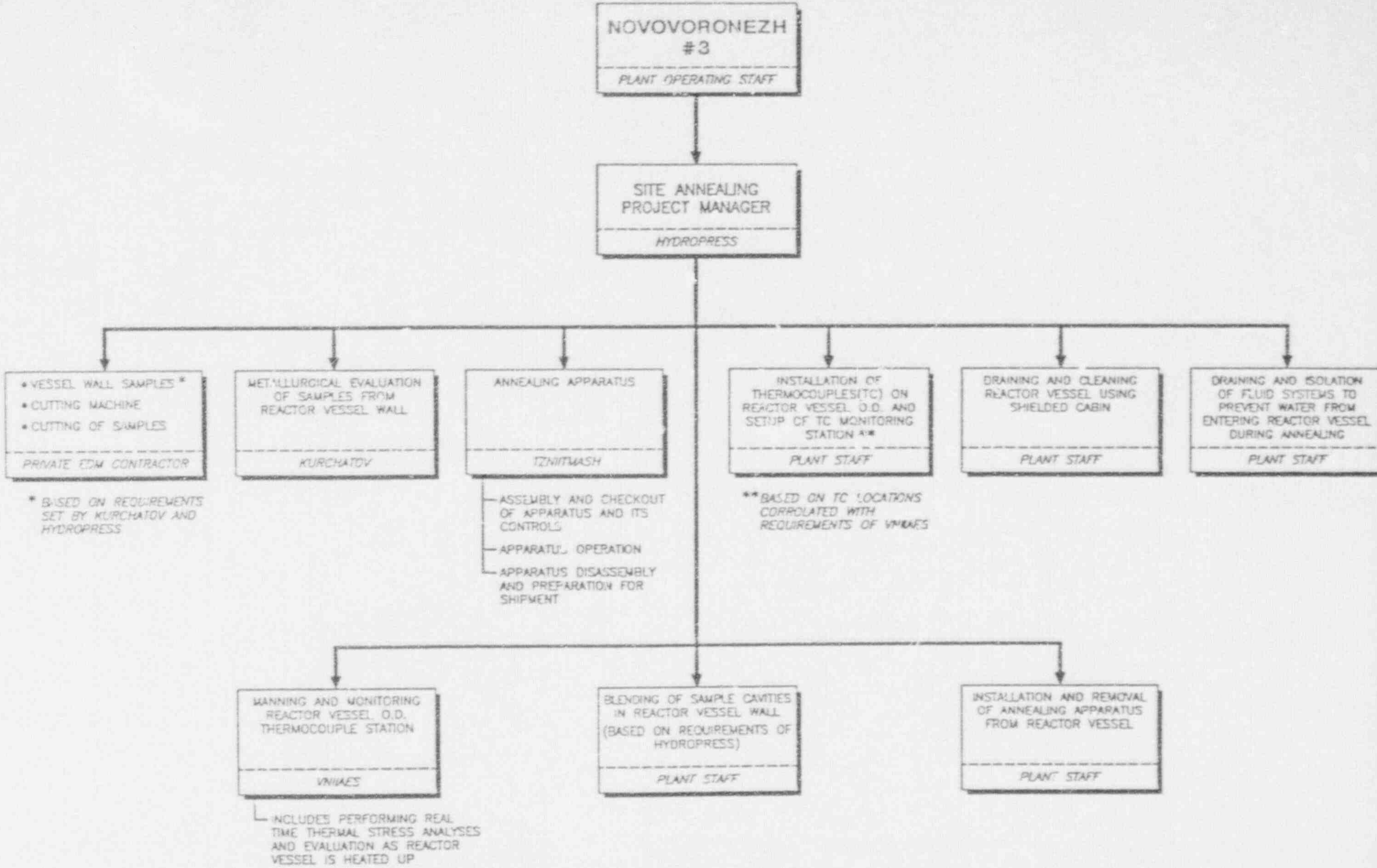
Noman M. Cole, Jr.
MPR Associates, Inc.

IX. FIGURES



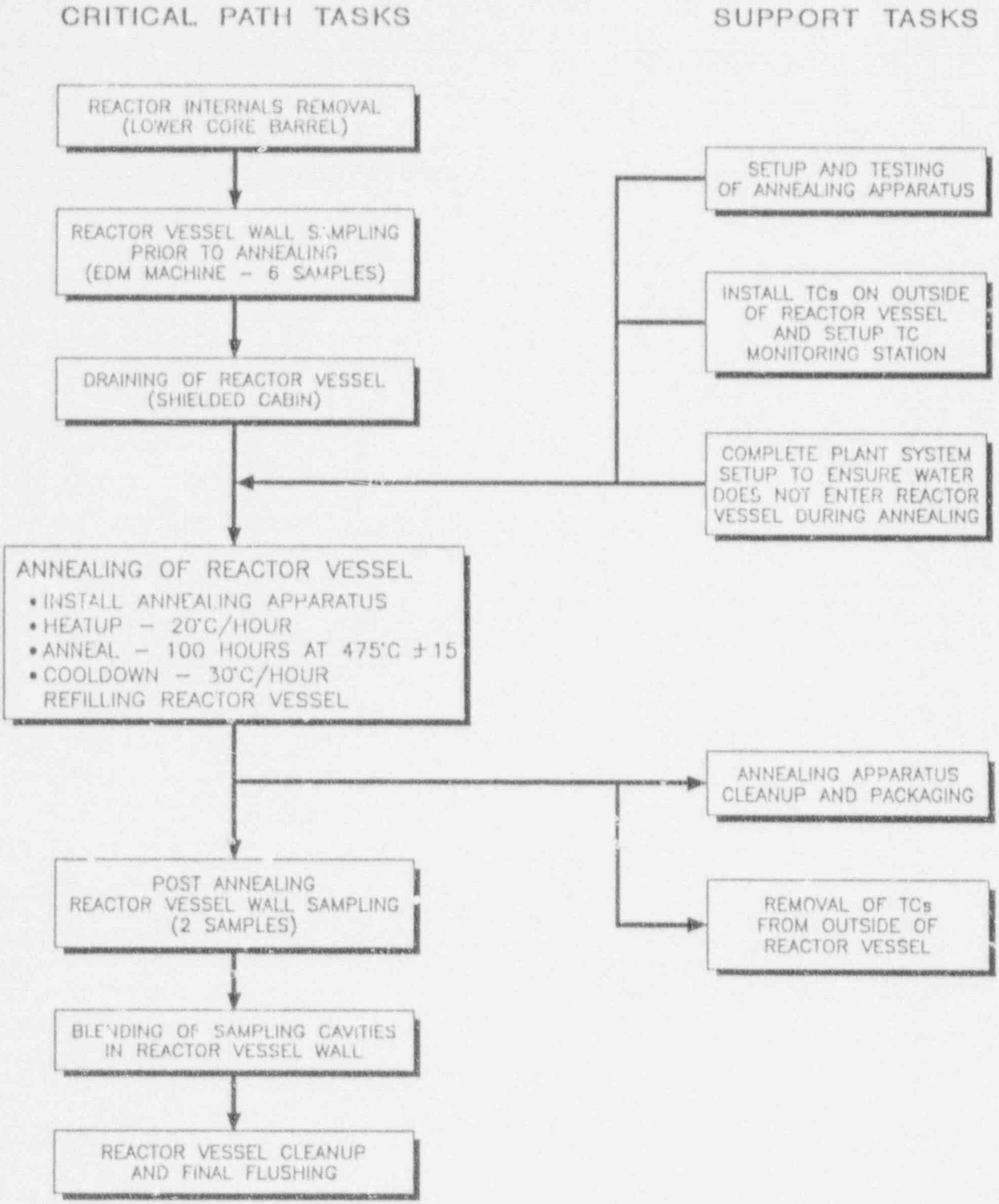
NOVOVORONEZH UNIT 3 AND UNIT 4
CENTRAL HALL AREA
GENERAL ARRANGEMENT

FIGURE 1



OVERALL ORGANIZATIONAL SETUP FOR SITE REACTOR VESSEL ANNEALING

FIGURE 2

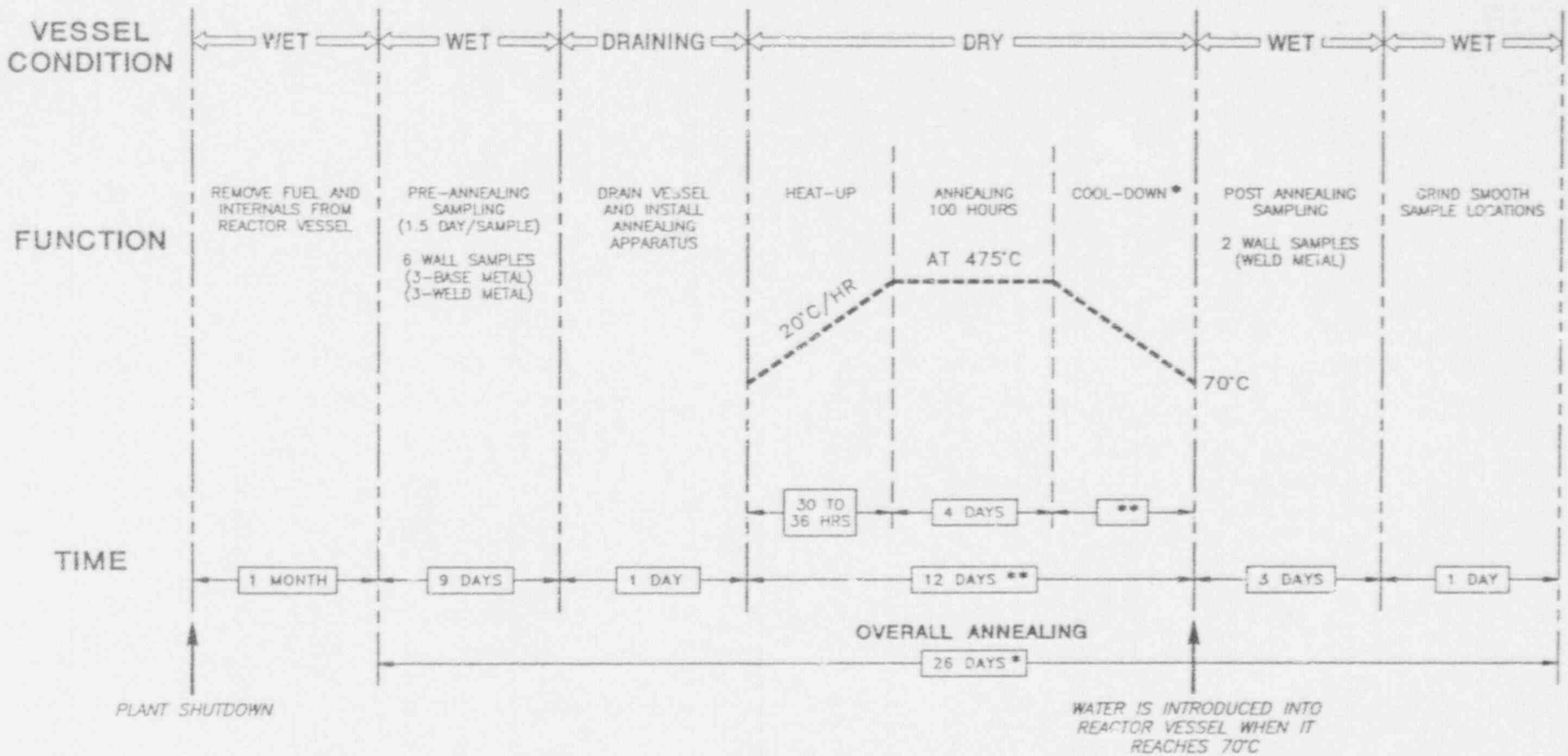


OVERALL ANNEALING SEQUENCE FOR NOVOVORONEZH #3
FEBRUARY/MARCH 1991

FIGURE 3

SCHEDULE FOR GENERAL PLANNING PURPOSES

SHOWING MAJOR STEPS IN OVERALL ANNEALING SEQUENCE AND APPROXIMATE TIMES, BASED ON AROUND-THE-CLOCK OPERATIONS

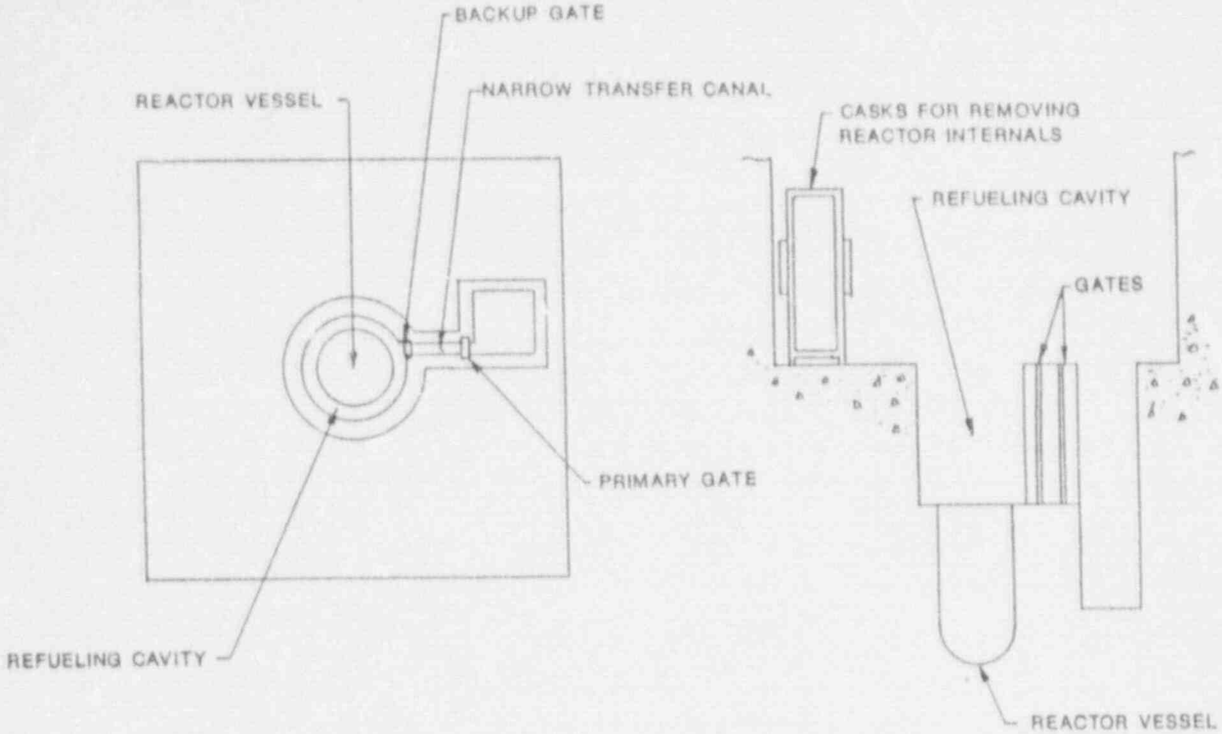


NOTES:

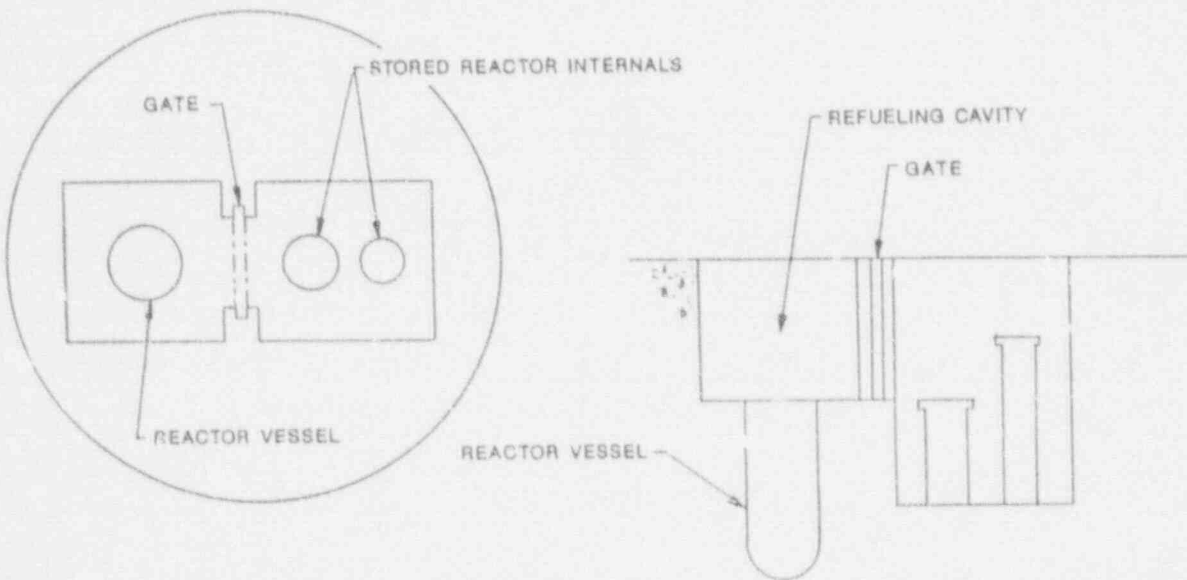
* COOL-DOWN LIMIT IS 30°C/HOUR; HOWEVER, BETWEEN 300°C AND 70°C IT TAKES MORE TIME TO COOL DOWN THAN DESIRED. SOVIETS INDICATED THAT THIS PHASE TYPICALLY REQUIRED ABOUT 6 DAYS AND THAT THEY ARE LOOKING AT WAYS TO IMPROVE COOL-DOWN RATE IN THIS TEMPERATURE RANGE.

** THE ACTUAL ANNEALING OPERATIONS (HEAT-UP, ANNEALING, AND COOL-DOWN AT NOVOVORONEZH UNIT 3 IN FEBRUARY/MARCH OF 1991 REQUIRED A TOTAL OF ABOUT 8 DAYS (200 HOURS). THE COOL-DOWN PHASE REQUIRED ABOUT 63 HOURS. THUS, THIS INDICATES THAT THEY WHERE SUCCESSFUL IN SAVING ABOUT 4 DAYS FROM THE OVERALL SCHEDULE.

FIGURE 4



VVER-440 PLANT

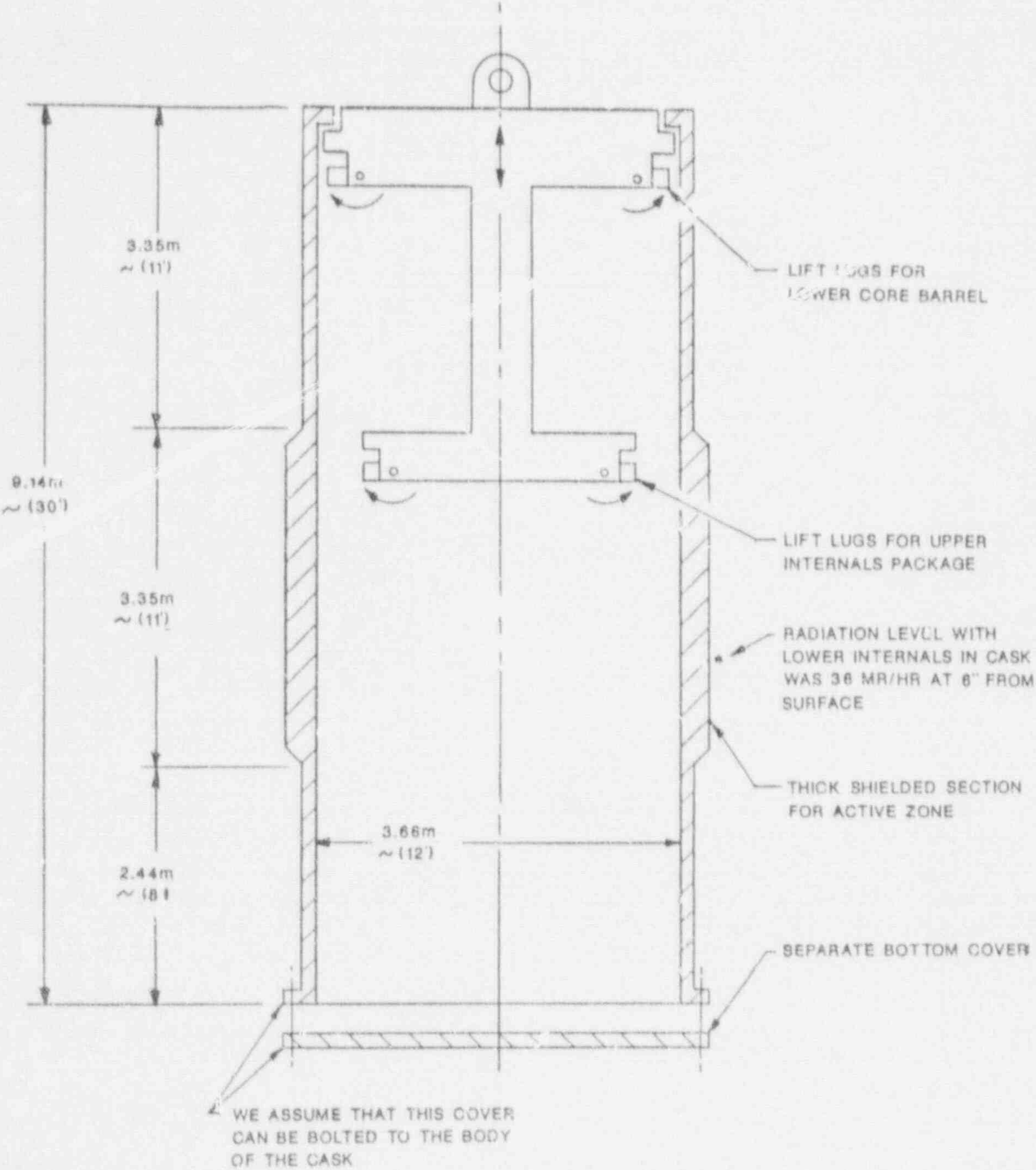


VVER-1000 PLANT

REACTOR VESSEL AND REFUELING CAVITY
CONCEPTUAL ARRANGEMENT

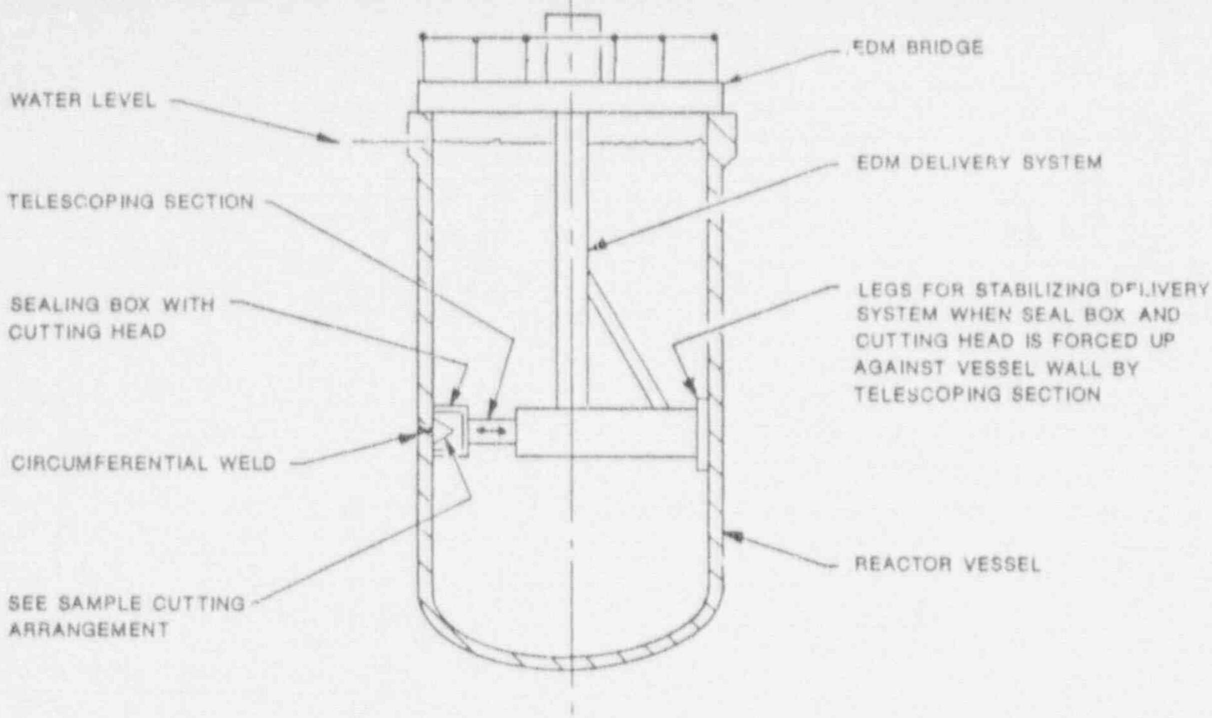
FIGURE 5

NOTE: COMBINED WEIGHT OF LOWER INTERNALS AND THE CASK IS ABOUT 277 US TONS

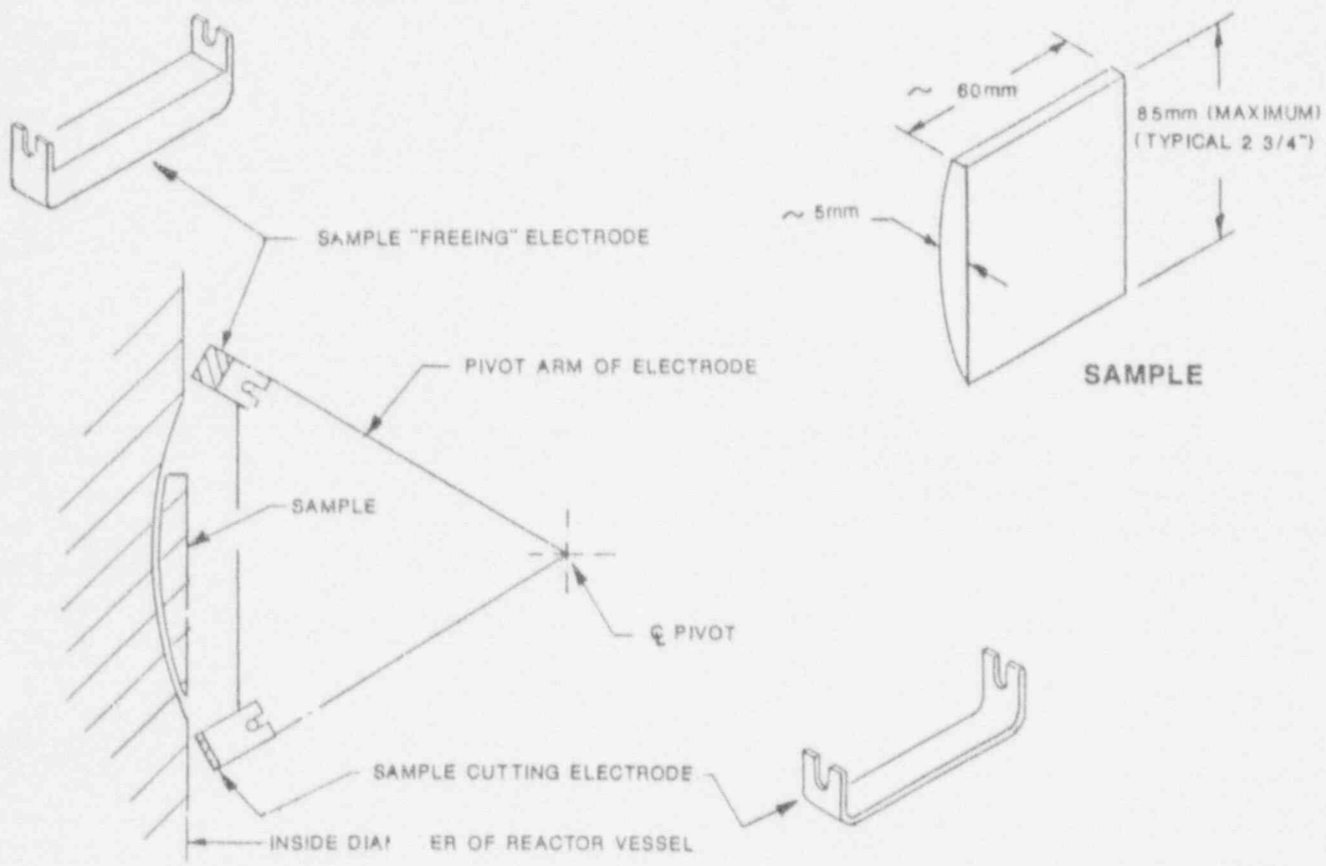


INTERNALS HANDLING CASK
CONCEPT

FIGURE 6



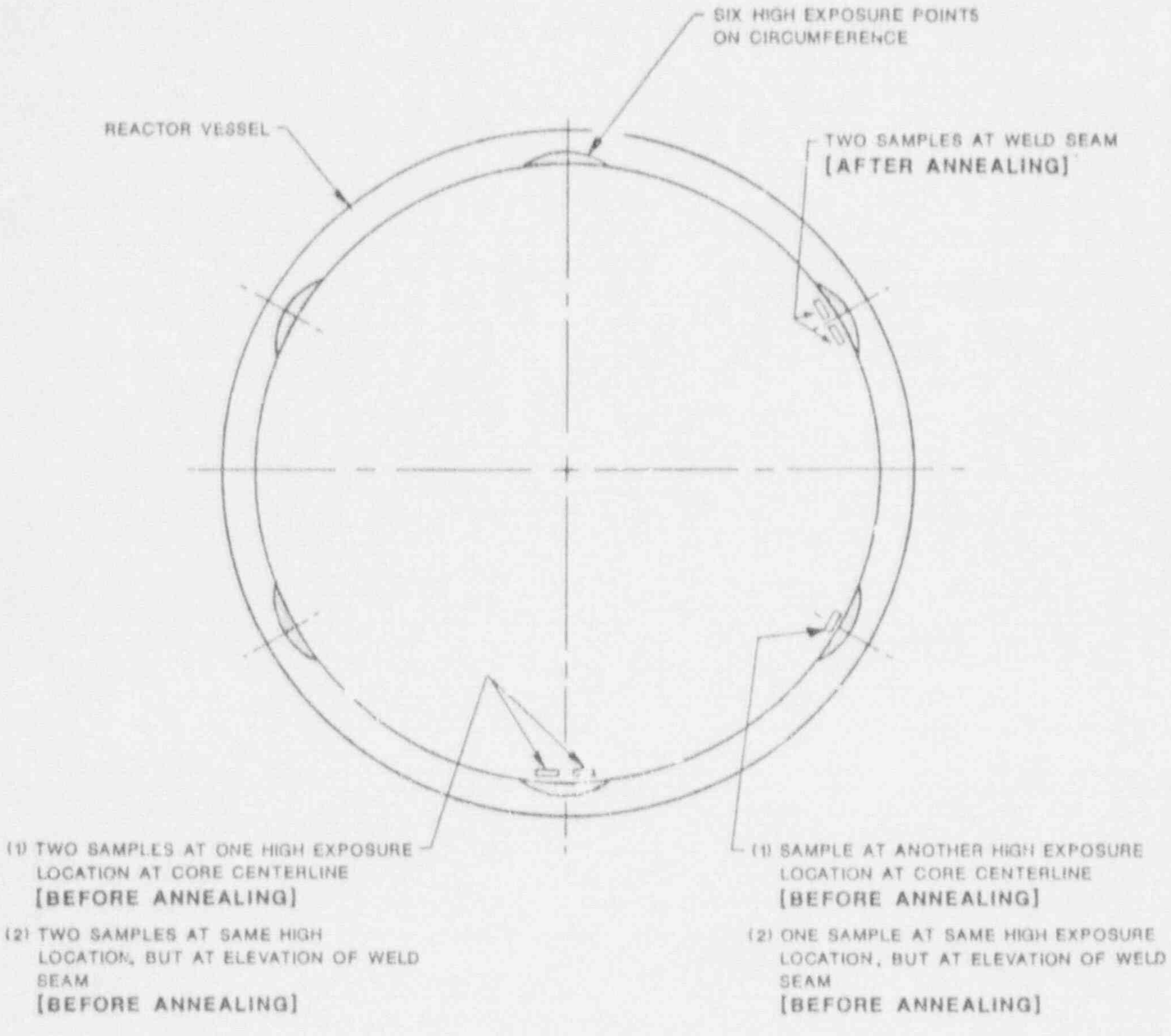
SECTION THRU REACTOR VESSEL



SAMPLE CUTTING ARRANGEMENT

EDM SAMPLE CUTTING MACHINE

FIGURE 7



REACTOR VESSEL WELD SAMPLES
CIRCUMFERENTIAL LOCATIONS

FIGURE 8

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

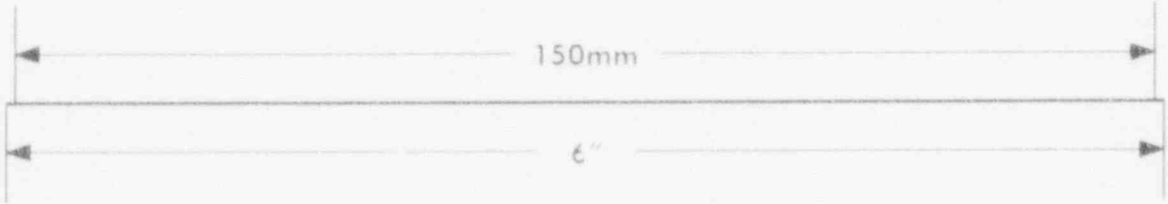
IMAGE EVALUATION TEST TARGET (MT-3)

150mm

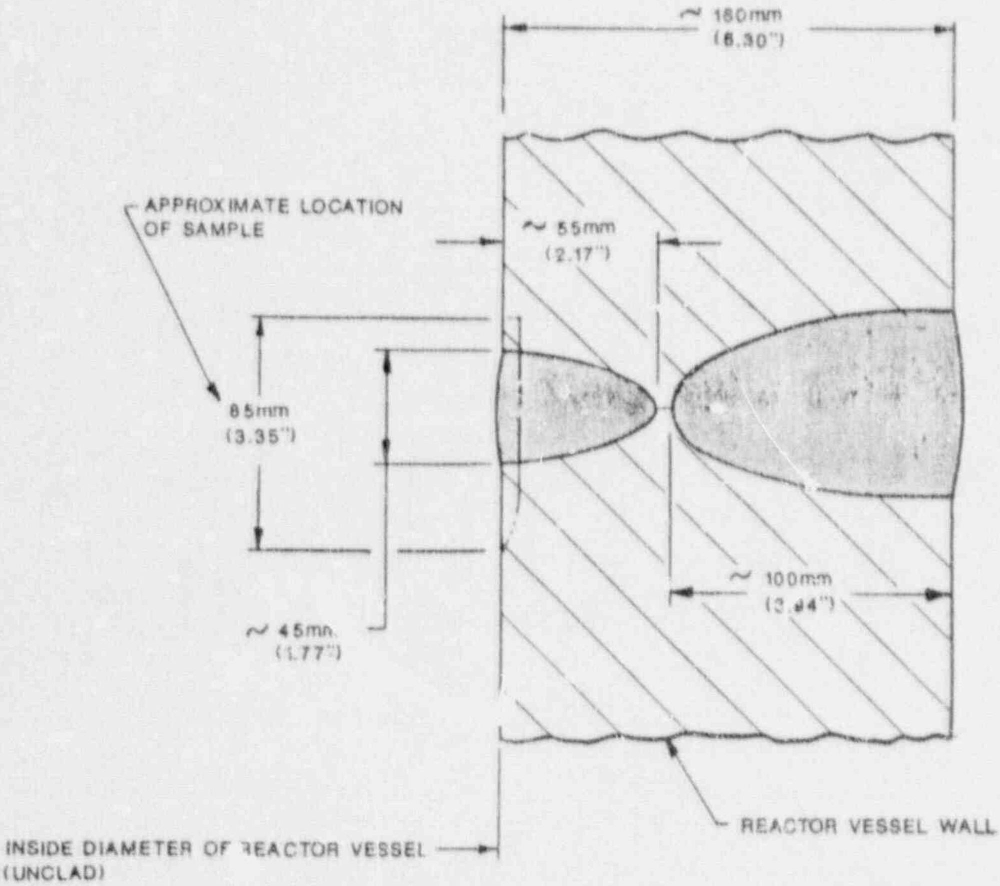
6"

1

IMAGE EVALUATION TEST TARGET (MT-3)



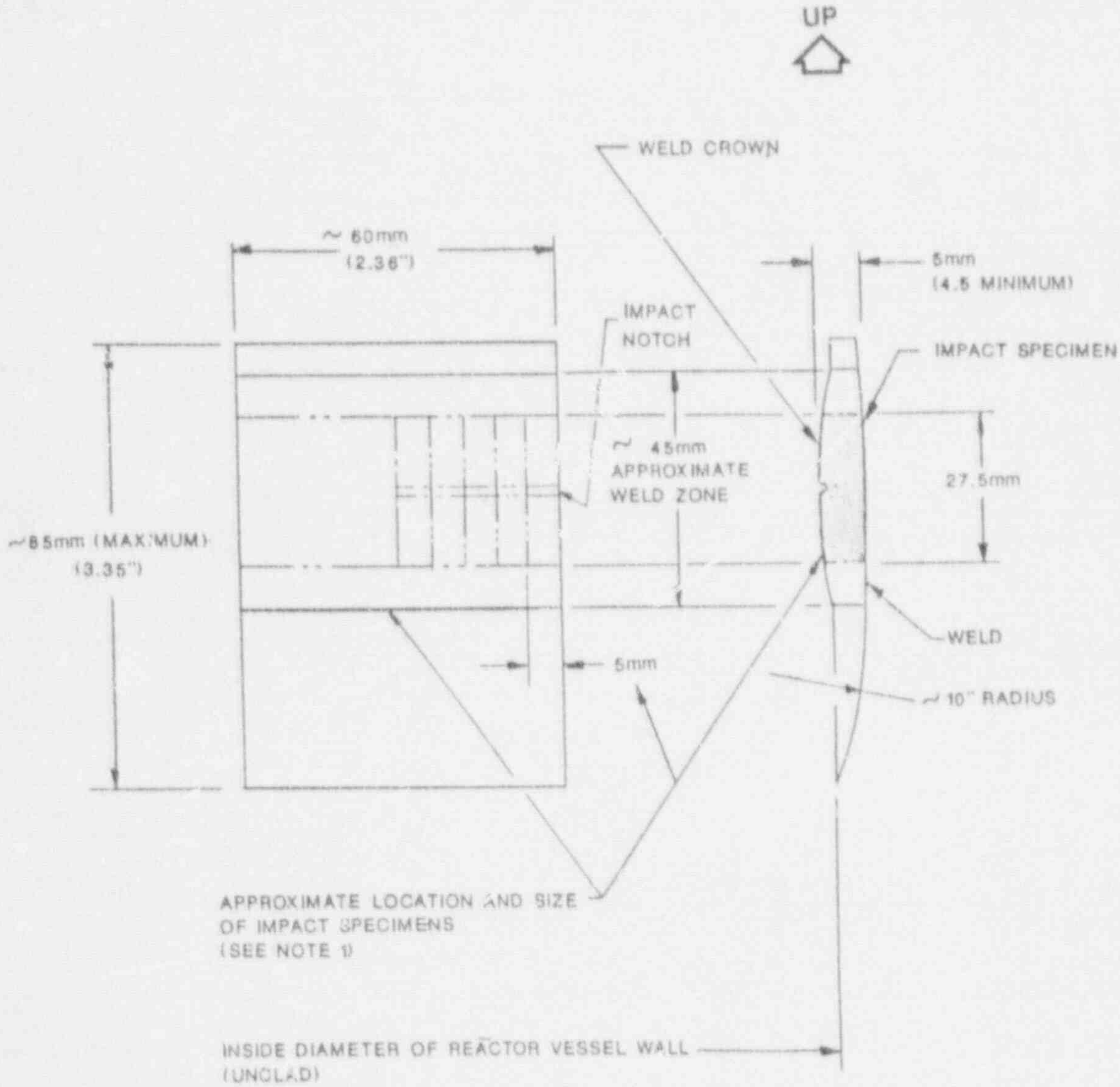
NOTE: ALL DIMENSIONS ARE APPROXIMATE



REACTOR VESSEL CIRCUMFERNTIAL WELD
DETAIL

FIGURE 9

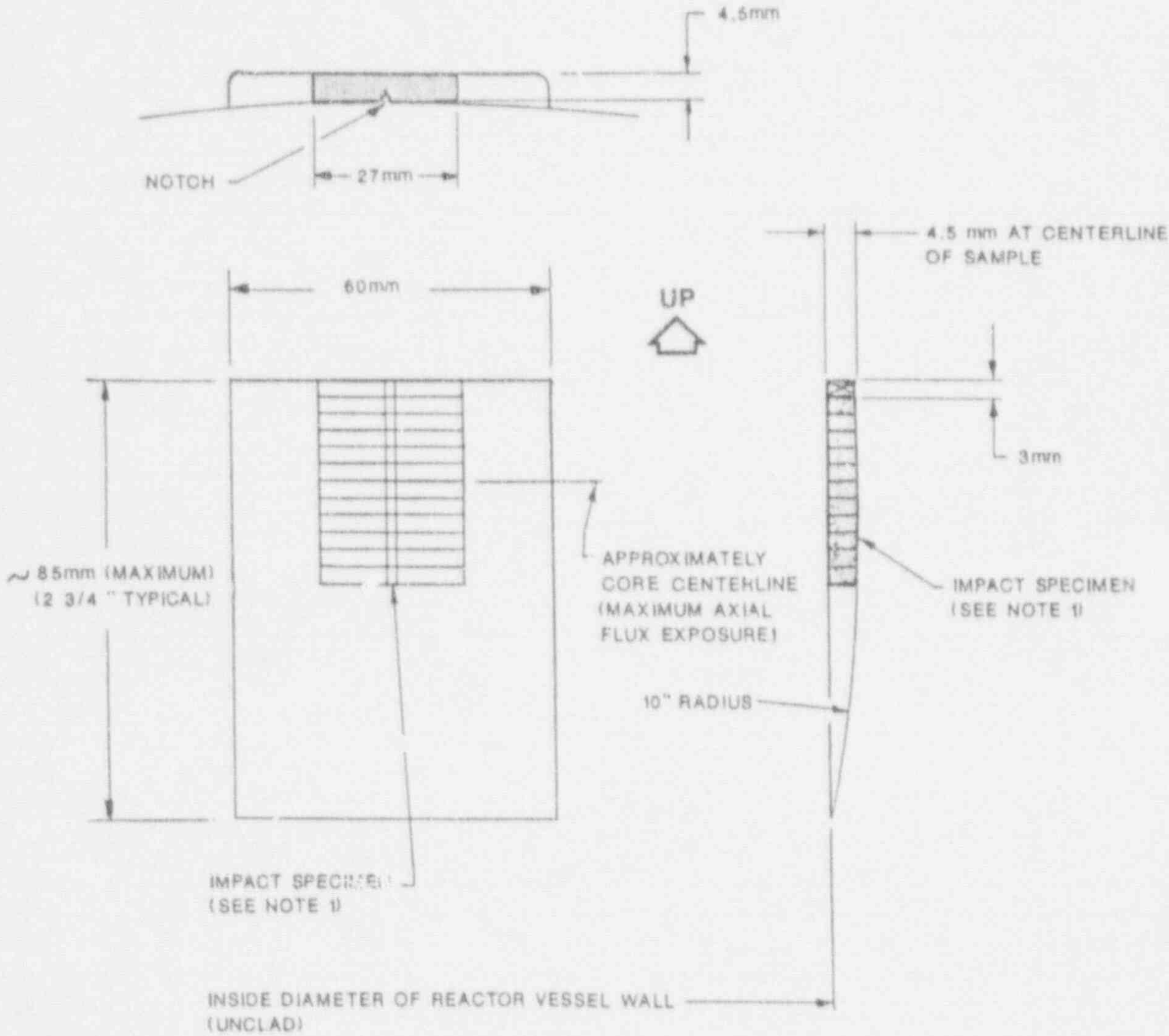
NOTE: APPROXIMATELY 9 TO 10 IMPACT SPECIMENS OBTAINED PER SAMPLE. IMPACT SPECIMENS IN WELD AREA ARE 5mm X 5mm X 27.5mm.



SIZE OF WELD SAMPLE REMOVED
FROM THE REACTOR VESSEL WALL
AND
SIZE AND ORIENTAION OF IMPACT SPECIMENS

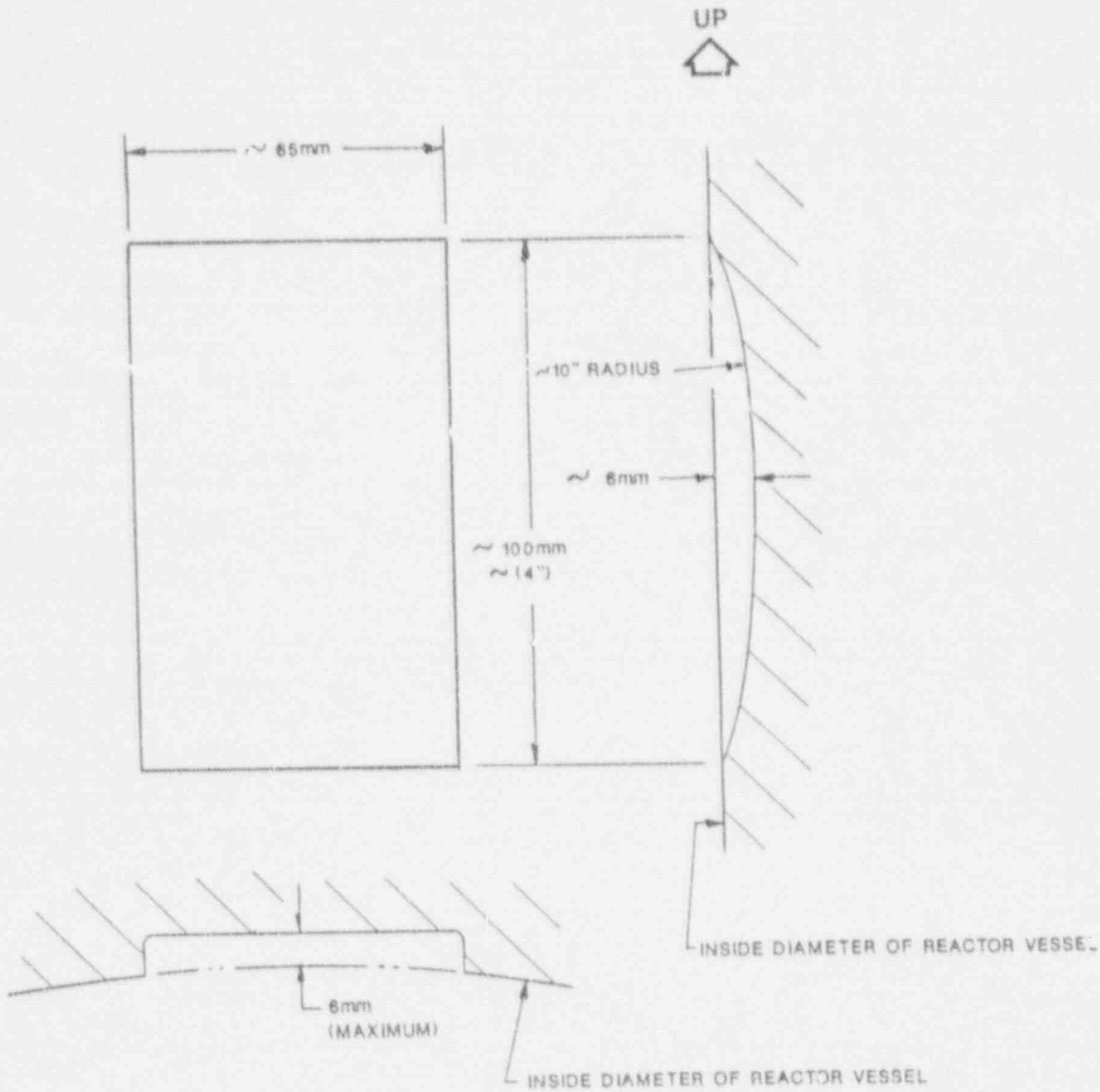
FIGURE 10

NOTE: 1. APPROXIMATELY 12 IMPACT SPECIMENS OBTAINED PER SAMPLE.
IMPACT SPECIMEN SIZE IS 4.5mm X 3mm X 27mm
(STANDARD SIZE USED BY GERMANS)



SIZE OF BASE METAL SAMPLE REMOVED
FROM THE REACTOR VESSEL WALL
AND
SIZE AND ORIENTATION OF IMPACT SPECIMENS

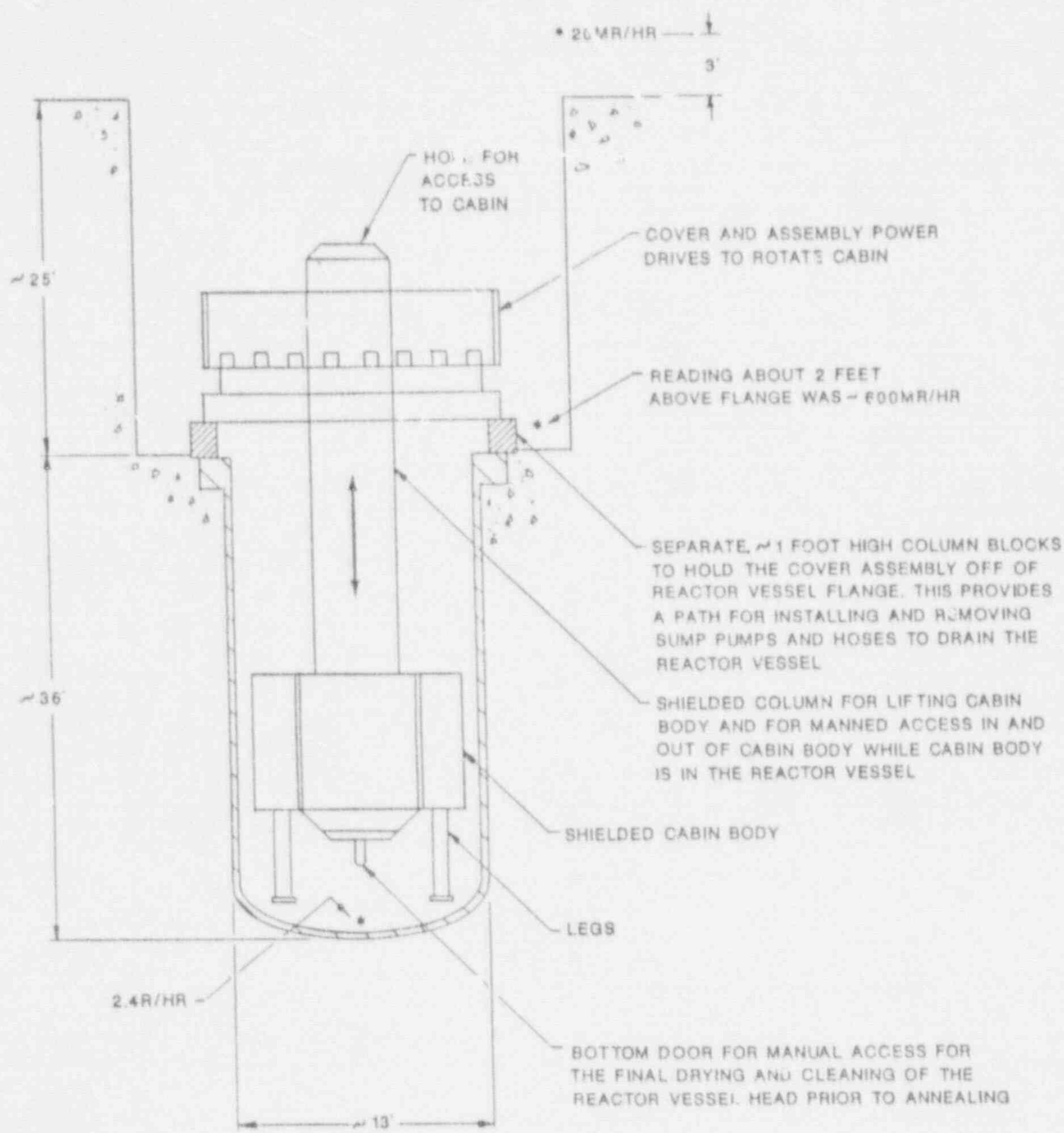
FIGURE 11



CAVITY SIZE IN REACTOR VESSEL WALL
DUE TO SAMPLE REMOVAL

FIGURE 12

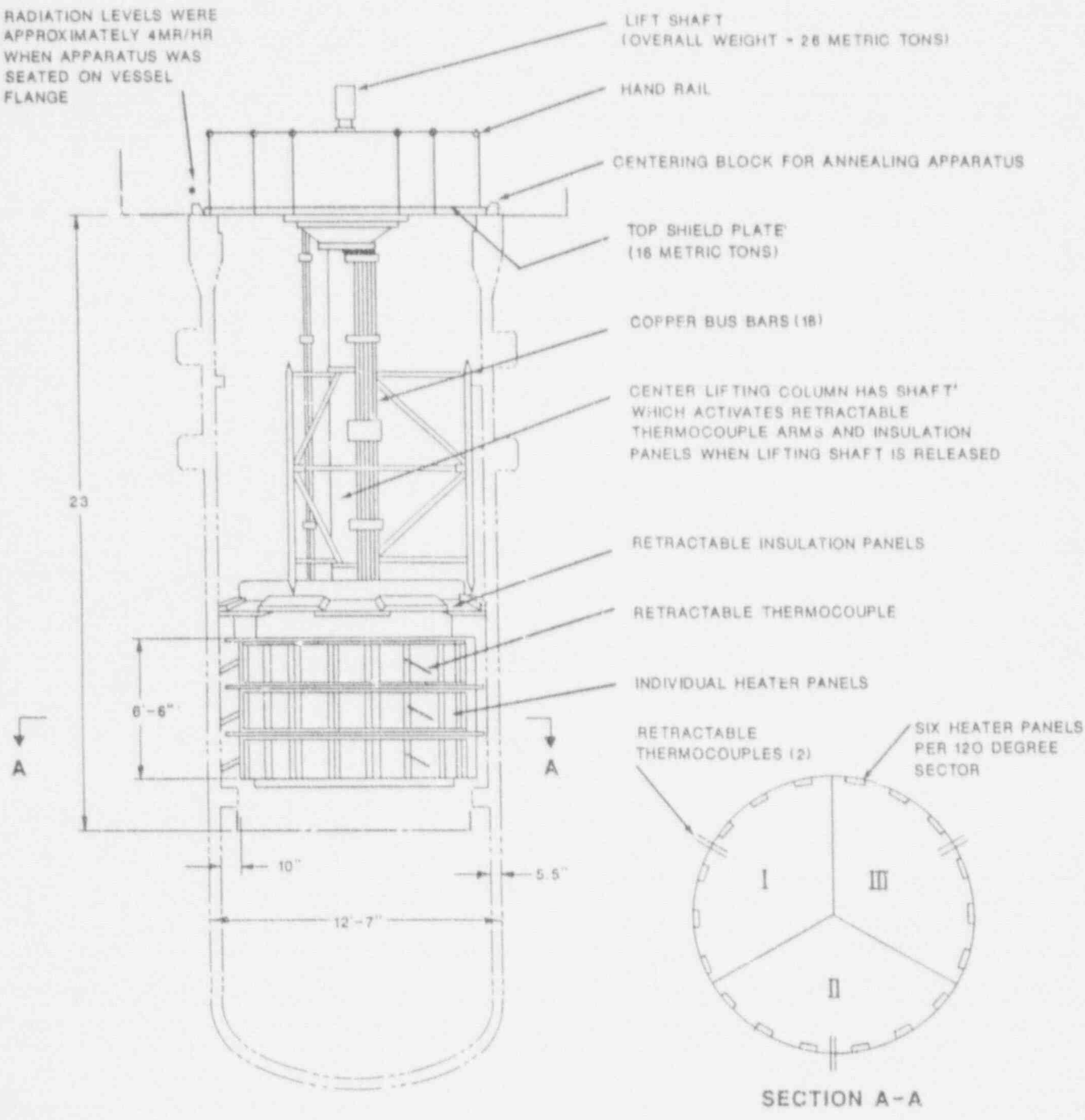
MEASURED RADIATION LEVELS AT NOVOVORENZH UNIT 3



SHIELDED CABIN IN REACTOR VESSEL
FINAL DRYING AND CLEANING OF REACTOR VESSEL

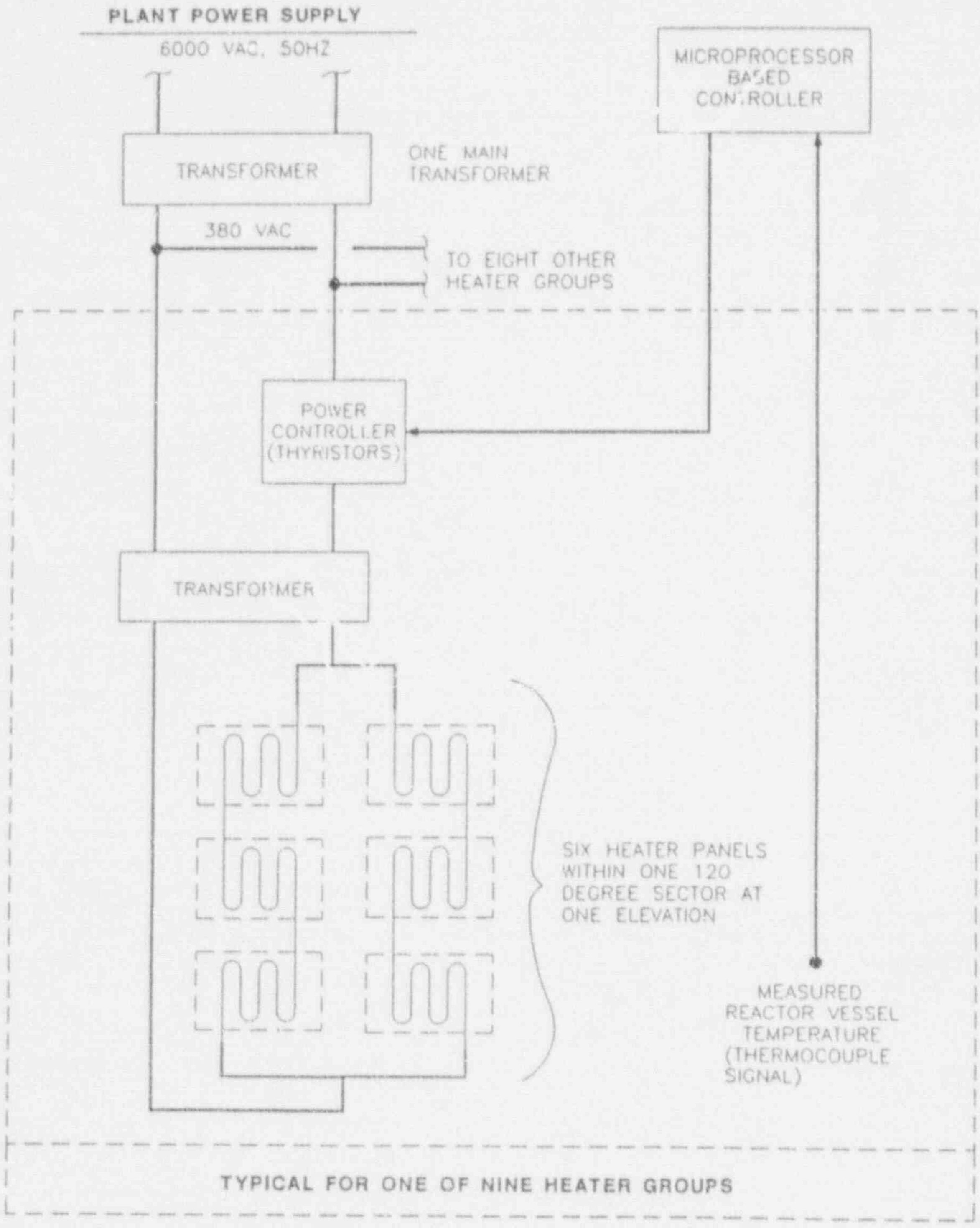
FIGURE 13

NOTE: ALL DIMENSIONS ARE APPROXIMATE



ANNEALING APPARATUS
(1 METER ANNEALING ZONE UNIT)
GENERAL ARRANGEMENT

FIGURE 14



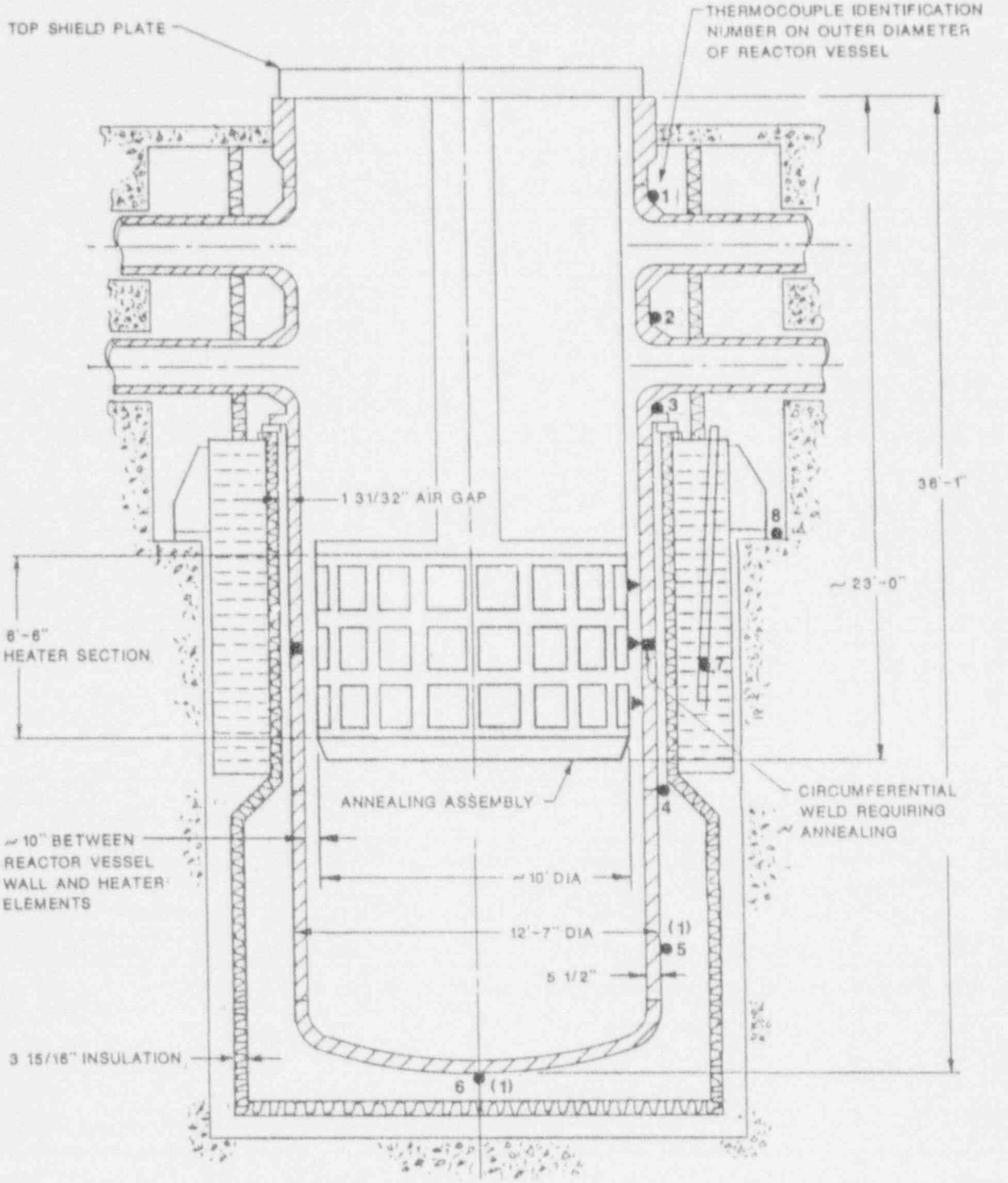
ANNEALING APPARATUS
ELECTRICAL CONTROL
SCHEMATIC

FIGURE 15

NOTE: 1. ▶ THERMOCOUPLES ON INSIDE DIAMETER OF REACTOR VESSEL
- SIX AT EACH ELEVATION - TWO THERMOCOUPLES IN EACH 120 DEGREE SECTOR (SEE FIGURE 10)

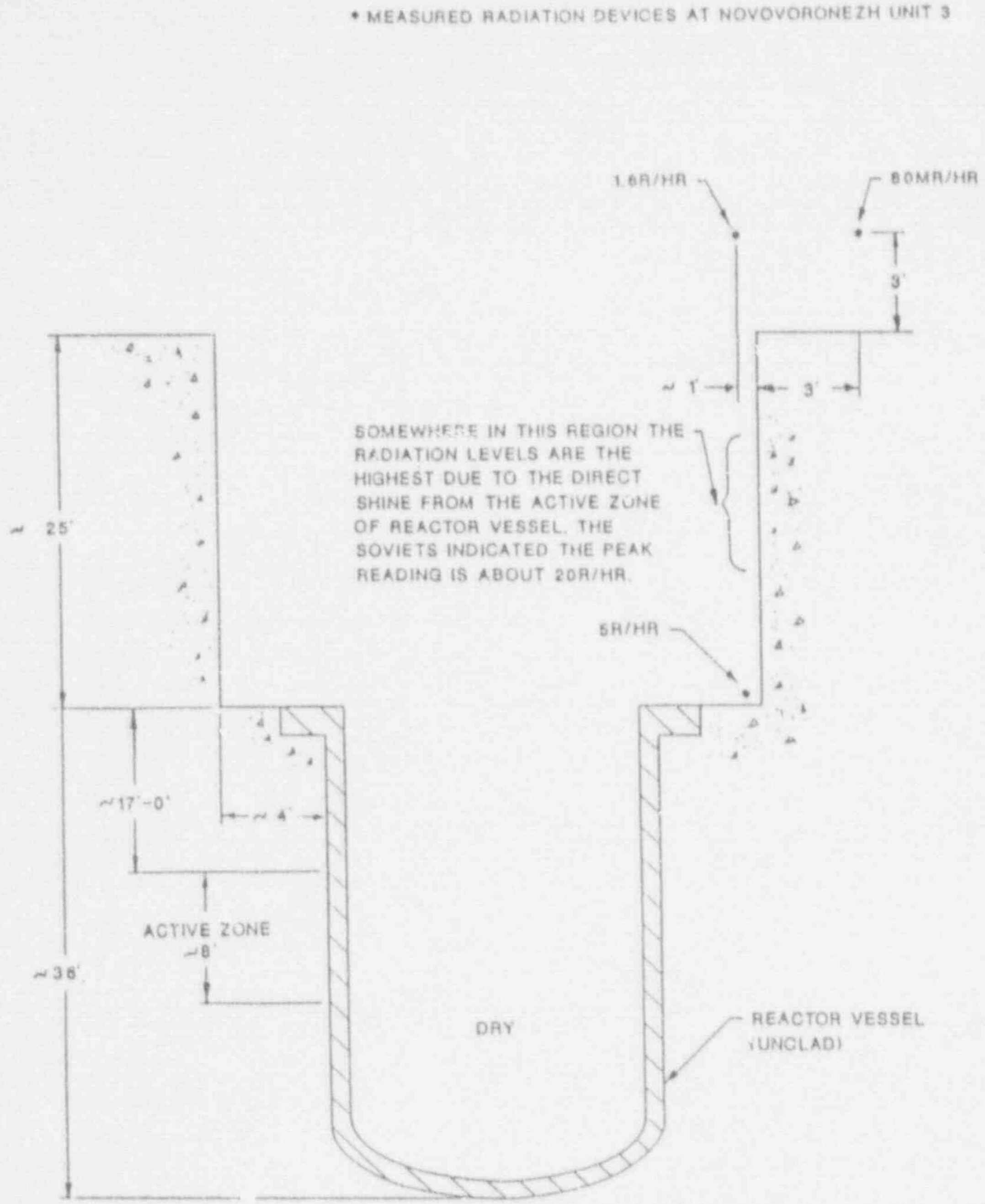
2. ● THERMOCOUPLES ON OUTSIDE DIAMETER OF REACTOR VESSEL
- THREE AT EACH ELEVATION, 20 DEGREES APART - EXCEPT AS SHOWN

3. ALL DIMENSIONS ARE APPROXIMATE



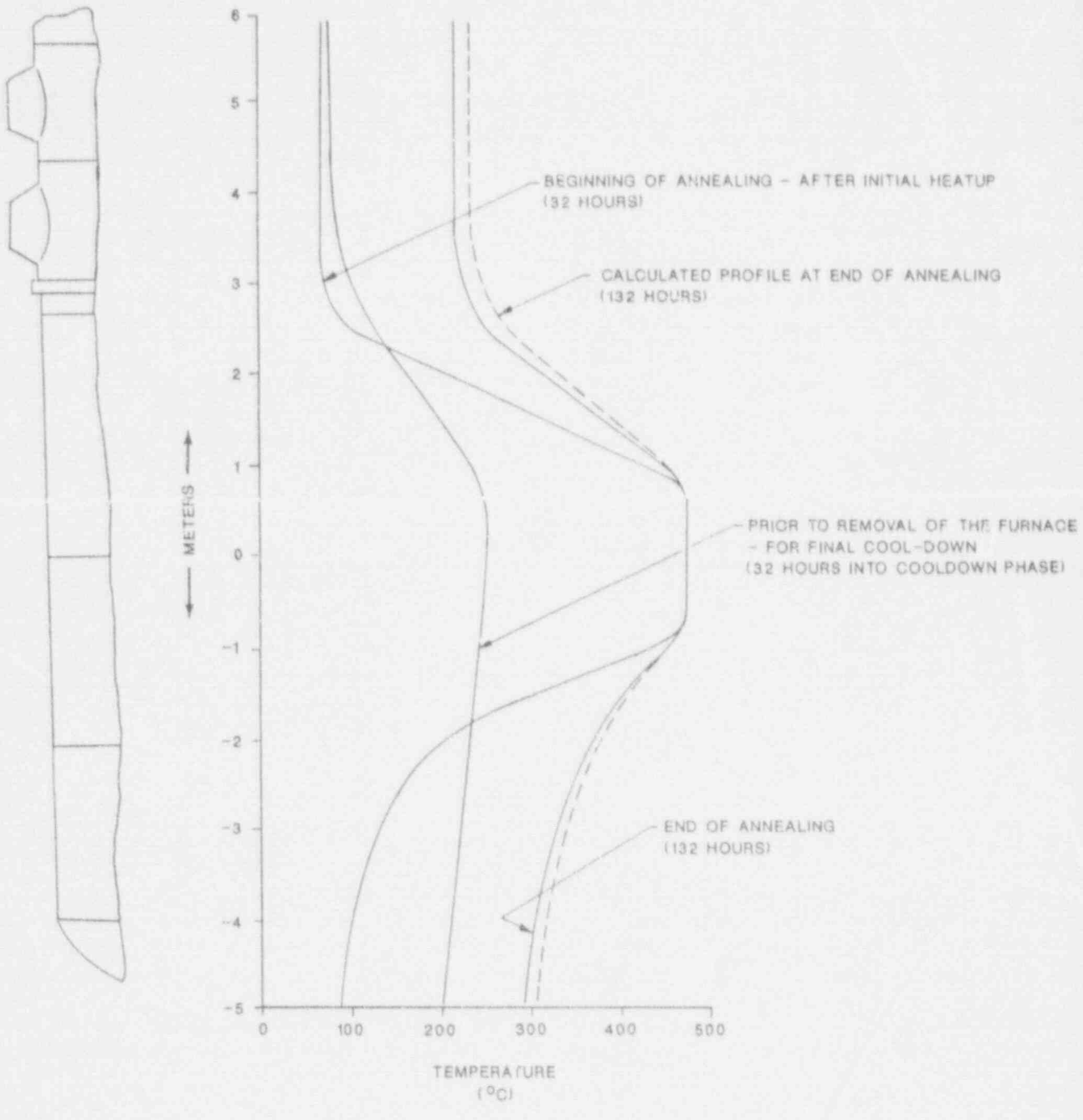
NOVOVORONEZH UNIT 3
THERMOCOUPLE LOCATIONS DURING
THE ANNEALING OF THE REACTOR VESSEL

FIGURE 16



RADIATION LEVELS
IN A DRY AND EMPTY REACTOR VESSEL
(UNCLAD VESSEL)

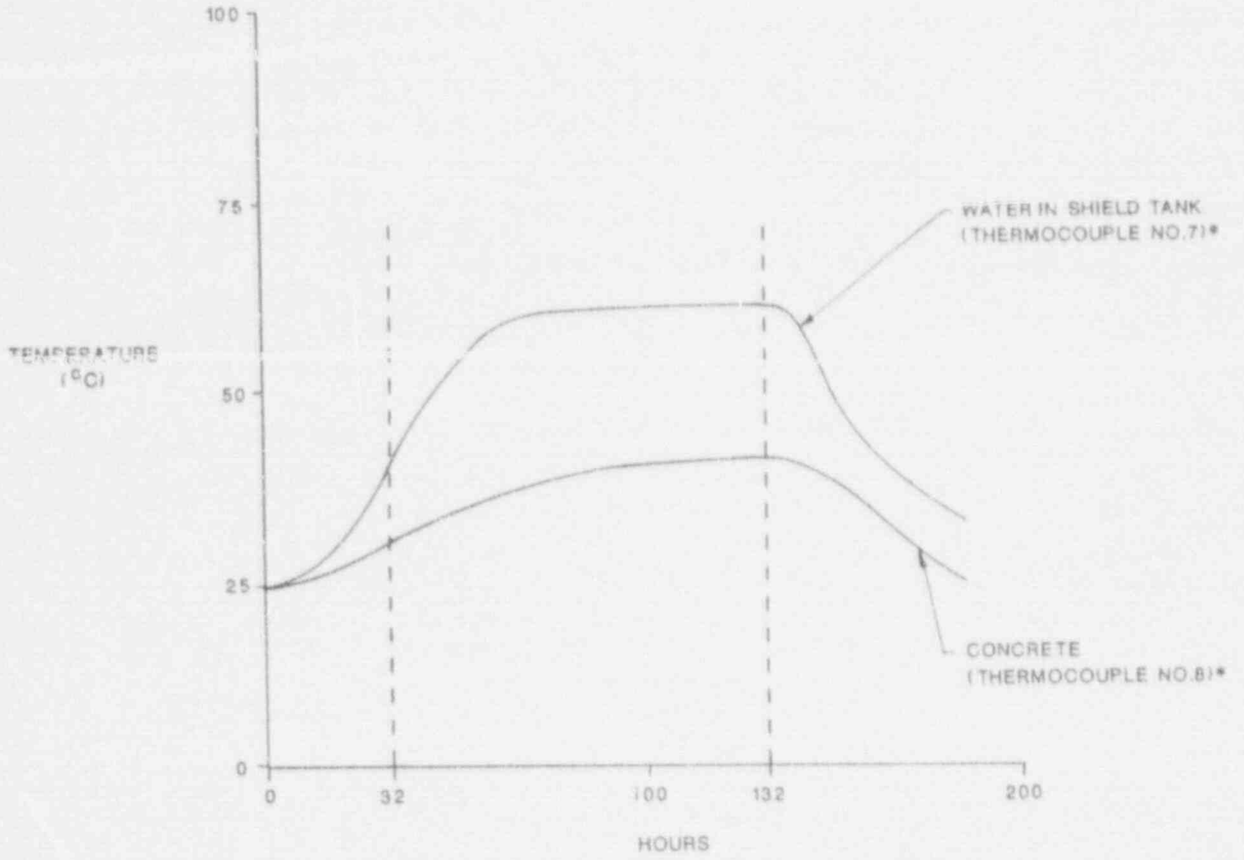
FIGURE 17



NOVOVORONEZH UNIT 3
ACTUAL REACTOR VESSEL WALL
TEMPERATURE PROFILES ALONG VESSEL HEIGHT
FEBRUARY - MARCH 1991

FIGURE 18

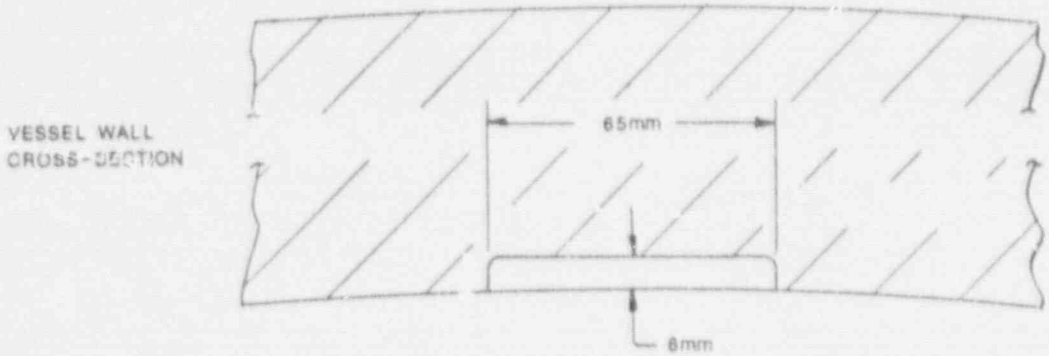
* SEE FIGURE 16



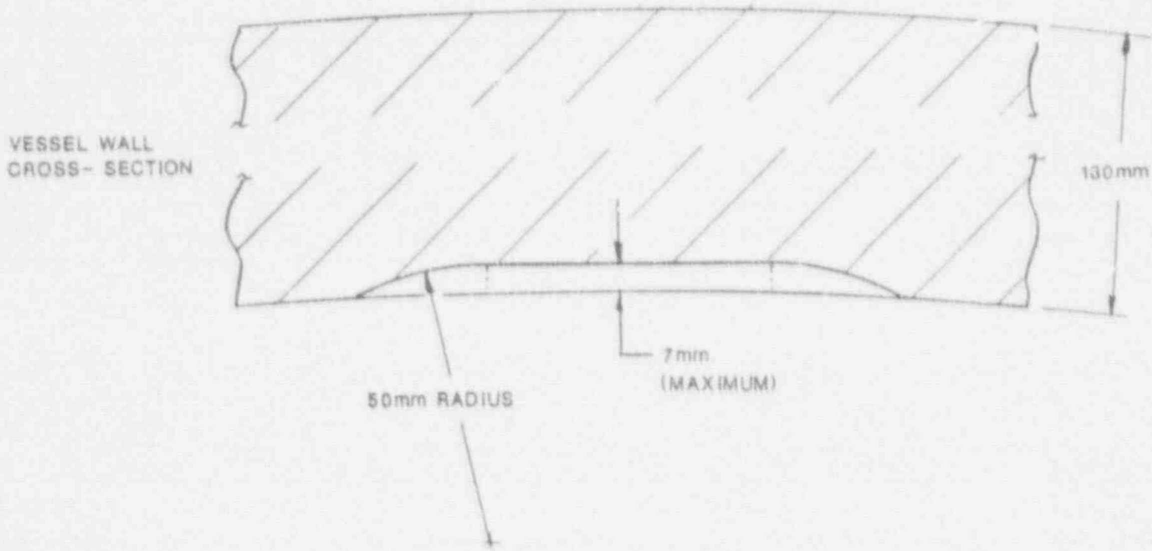
NOVOVORONEZH UNIT 3
CHANGES IN TEMPERATURE OF WATER
IN THE BIOLOGICAL SHIELD TANK AND IN
THE SUPPORTING CONCRETE DURING ANNEALING
FEBRUARY – MARCH 1991

FIGURE 19

NOTE: THE CURVATURE OF THE SAMPLE CAVITY IN THE VERTICAL ORIENTATION HAS A SMOOTH TRANSITION AND DOES NOT REQUIRE BLENDING (SEE FIGURE 6 AND FIGURE 8)



BEFORE BLENDING

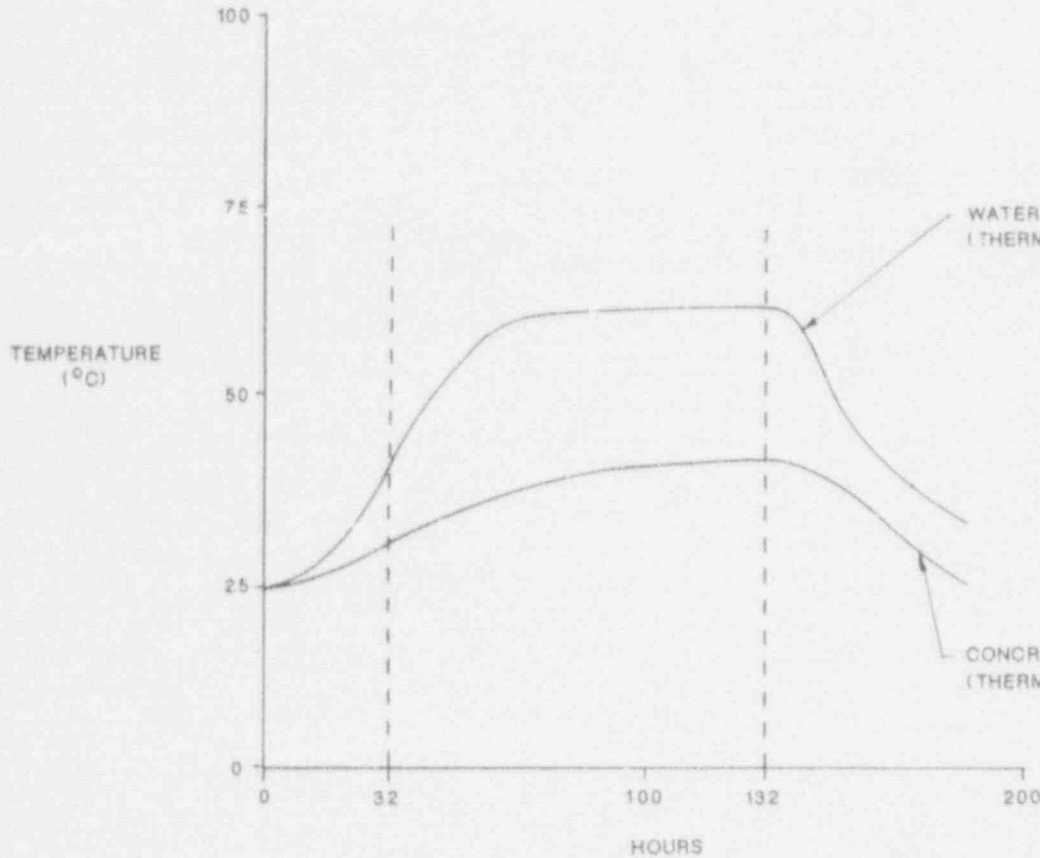


AFTER BLENDING

BLENDING SAMPLE CAVITIES IN REACTOR VESSEL WALL

FIGURE 20

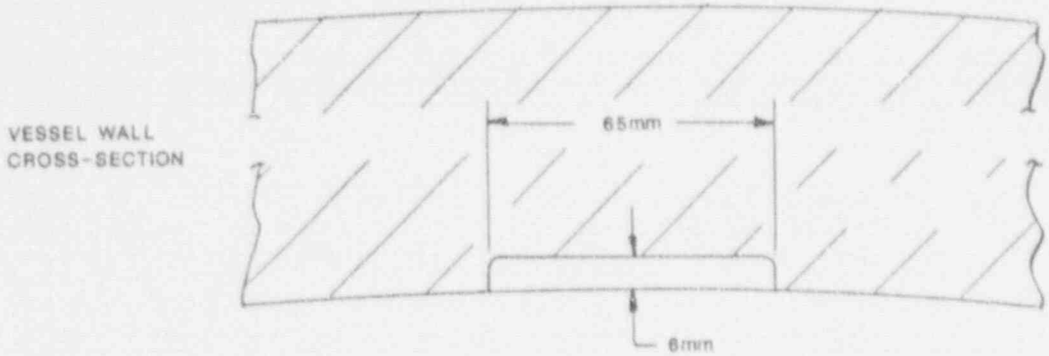
* SEE FIGURE 16



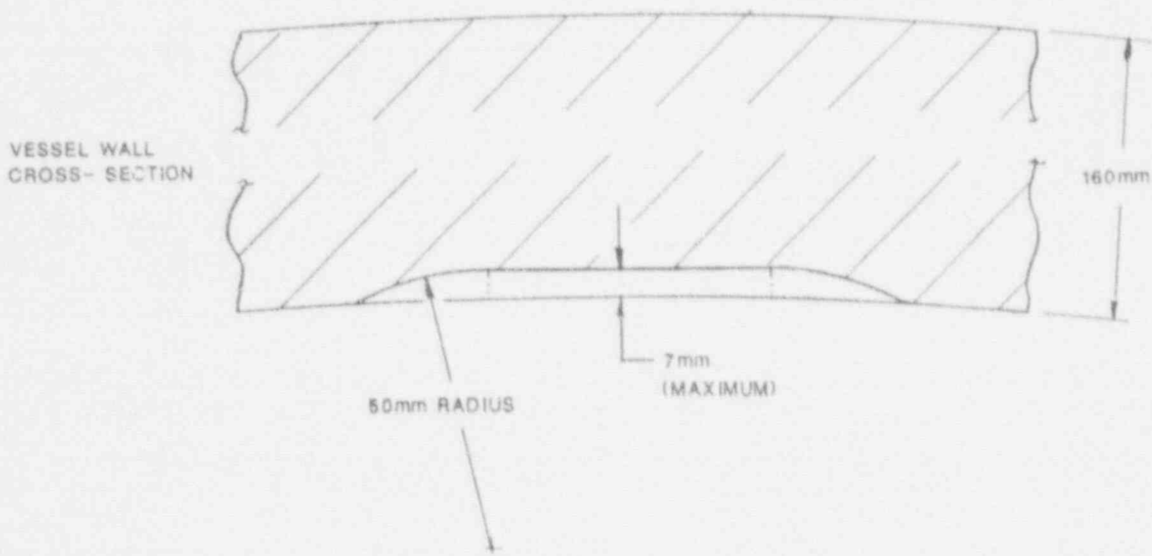
NOVOVORONEZH UNIT 3
CHANGES IN TEMPERATURE OF WATER
IN THE BIOLOGICAL SHIELD TANK AND IN
THE SUPPORTING CONCRETE DURING ANNEALING
FEBRUARY - MARCH 1991

FIGURE 19

NOTE: THE CURVATURE OF THE SAMPLE CAVITY IN THE VERTICAL ORIENTATION HAS A SMOOTH TRANSITION AND DOES NOT REQUIRE BLENDING (SEE FIGURE 6 AND FIGURE 8)

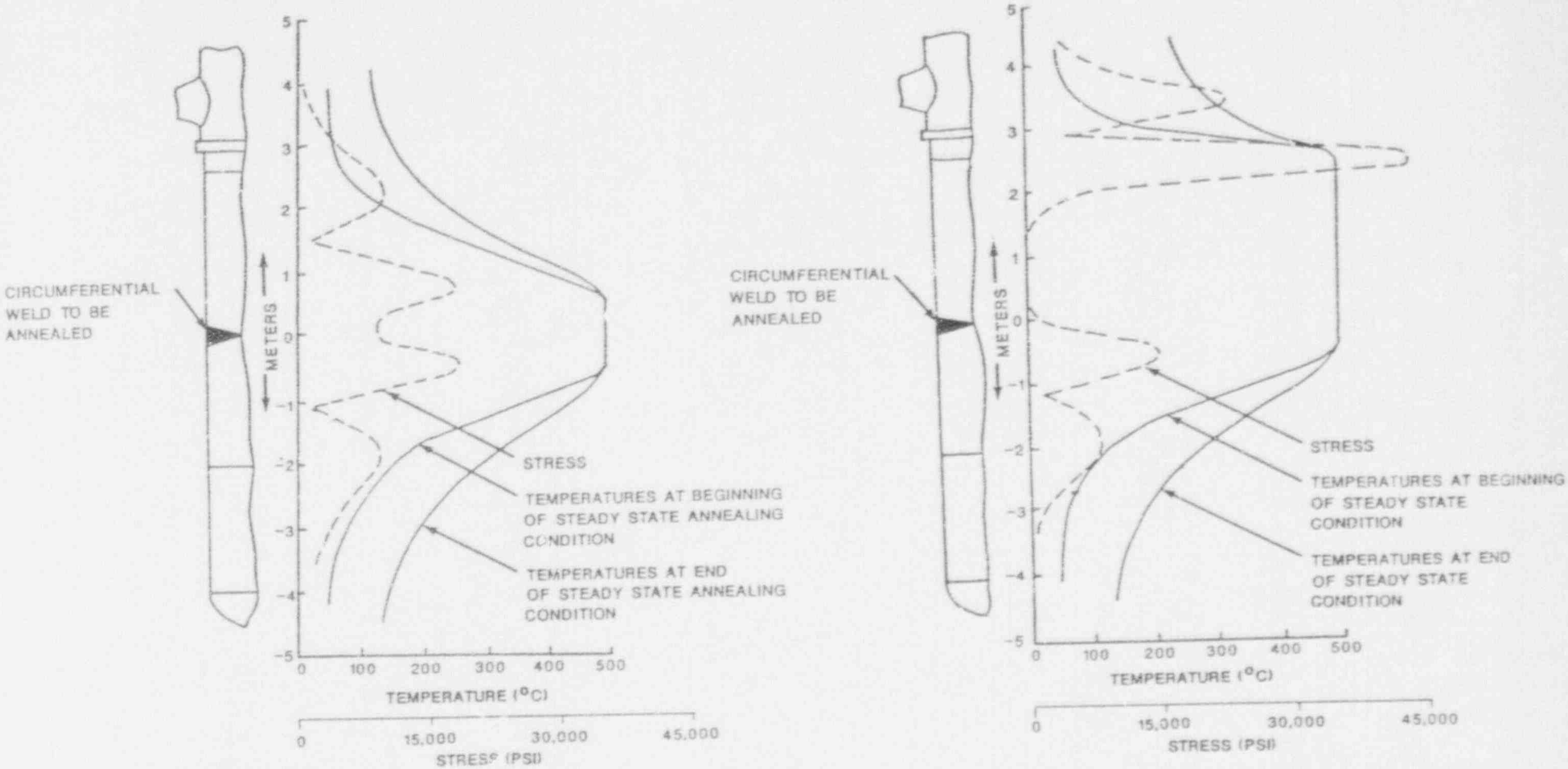


BEFORE BLENDING



AFTER BLENDING

BLENDING SAMPLE CAVITIES IN REACTOR VESSEL WALL
FIGURE 20

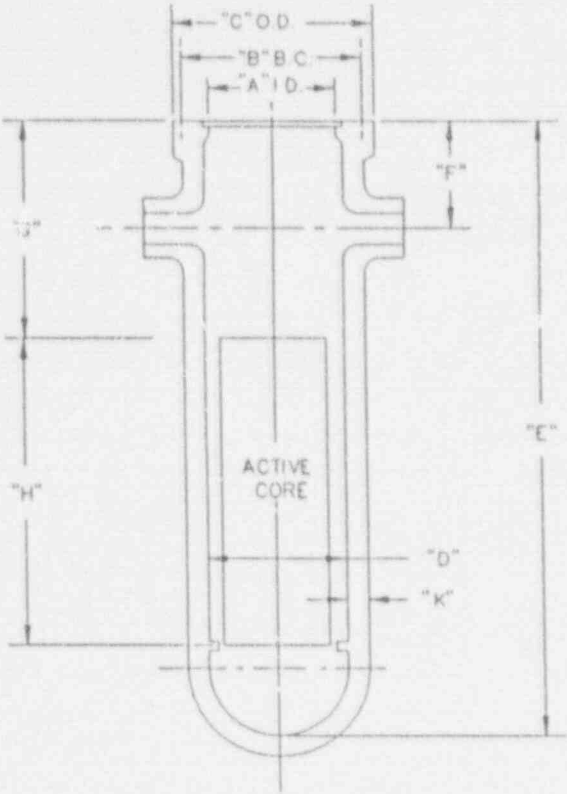


VVER 440 REACTOR VESSEL TEMPERATURE AND STRESS LEVELS DURING ANNEALING WITH **1 METER ANNEALING ZONE** HEATING APPARATUS

VVER 440 REACTOR VESSEL TEMPERATURE AND STRESS LEVELS DURING ANNEALING WITH **3 METER ANNEALING ZONE** HEATING APPARATUS

COMPARISON OF 1 METER AND 3 METER ANNEALING ZONE FOR VVER-440 REACTOR VESSEL

FIGURE 21



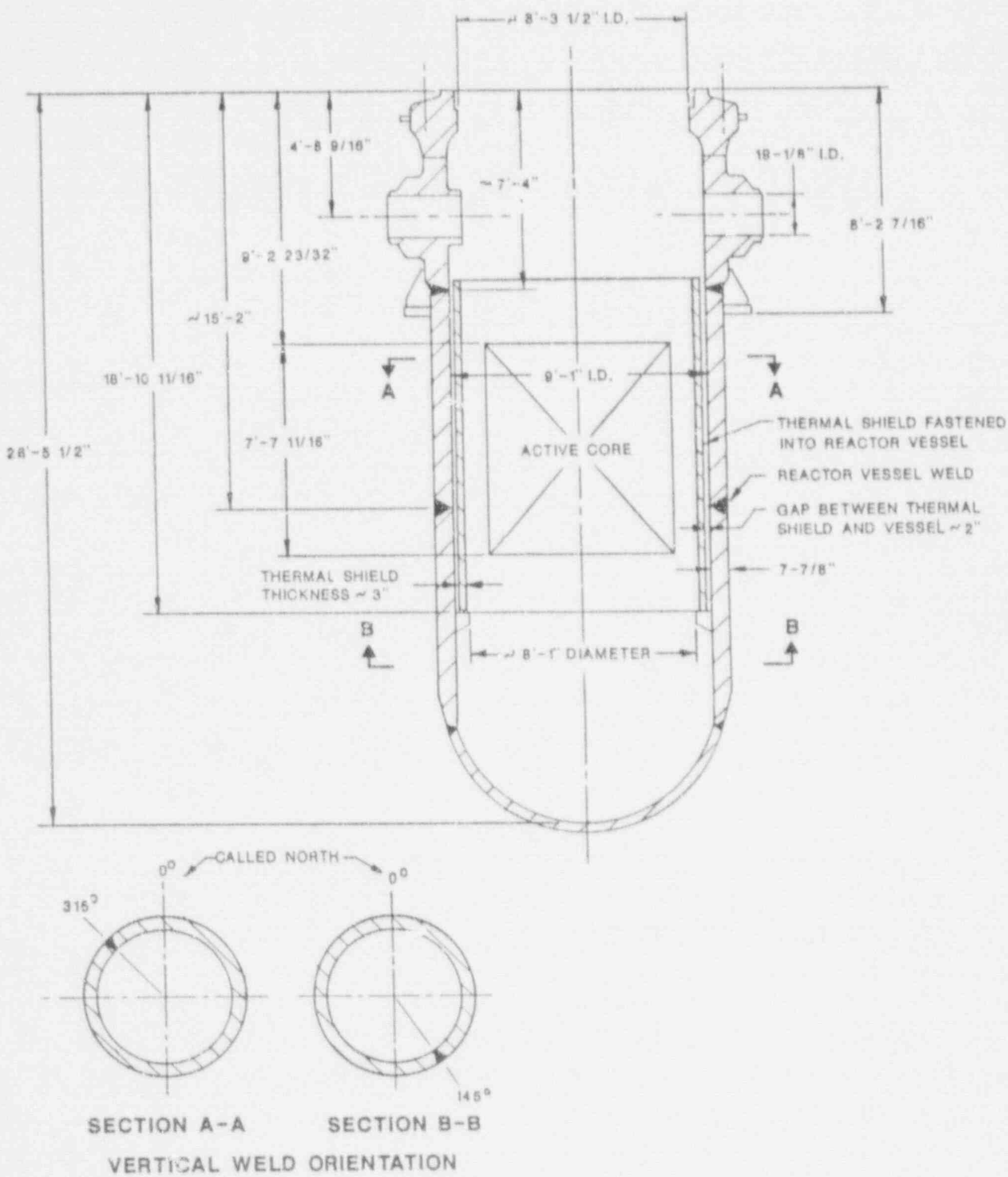
DIM	A	B	C	D	E	F	G	H	K *	CLAD THK	TYPE OF PLANT
CALVERT CLIFFS	166.00	192.13	205.00	172.00	403.88	80.50		136.70	8.62	.31	CE
TMI-2	164.88	186.75	200.00	170.62	379.00	84.00	138.00		8.44	.18	B&W
BELLEFONTE	178.00		217.25	182.00	396.00	85.50	138.50	143.00	9.25	.18	B&W
IPP	167.00	191.88	205.00	173.00	413.66	85.50	148.13	144.00			W 4-LOOP
TURKEY POINT	149.56	171.25	184.00	155.50	401.06	82.44			7.75	.16	W 3-LOOP

* Base Metal Only

Dimensions Are In Inches

TYPICAL U.S. PWR VESSELS

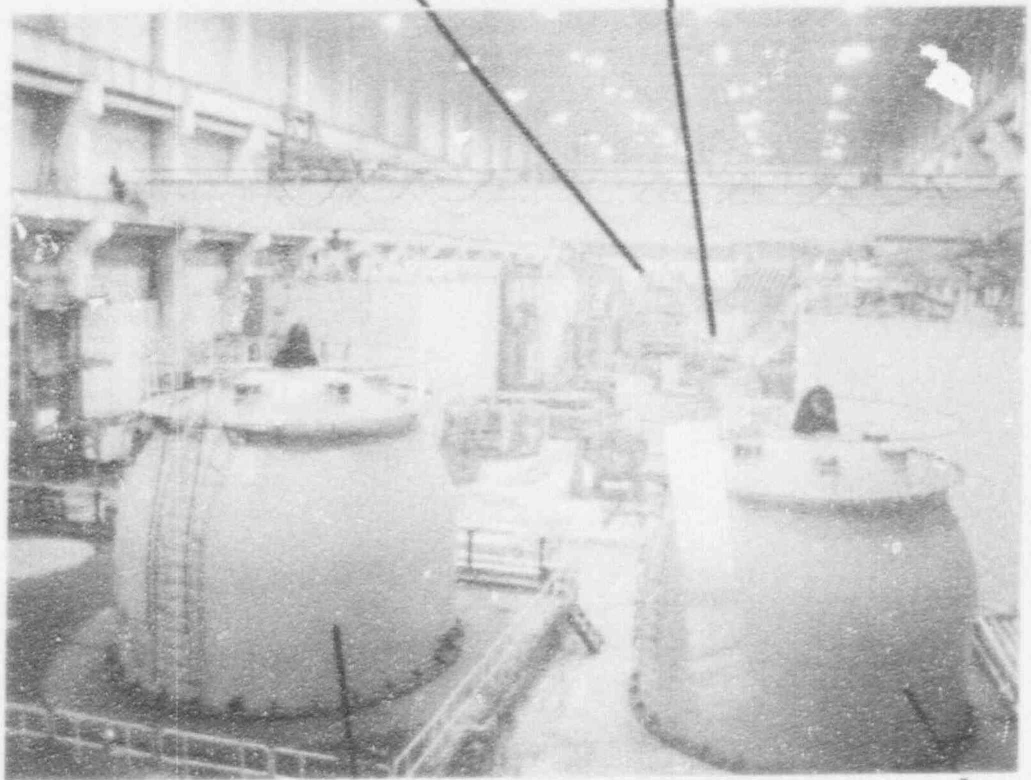
FIGURE 22



YANKEE ROWE
175 MWe

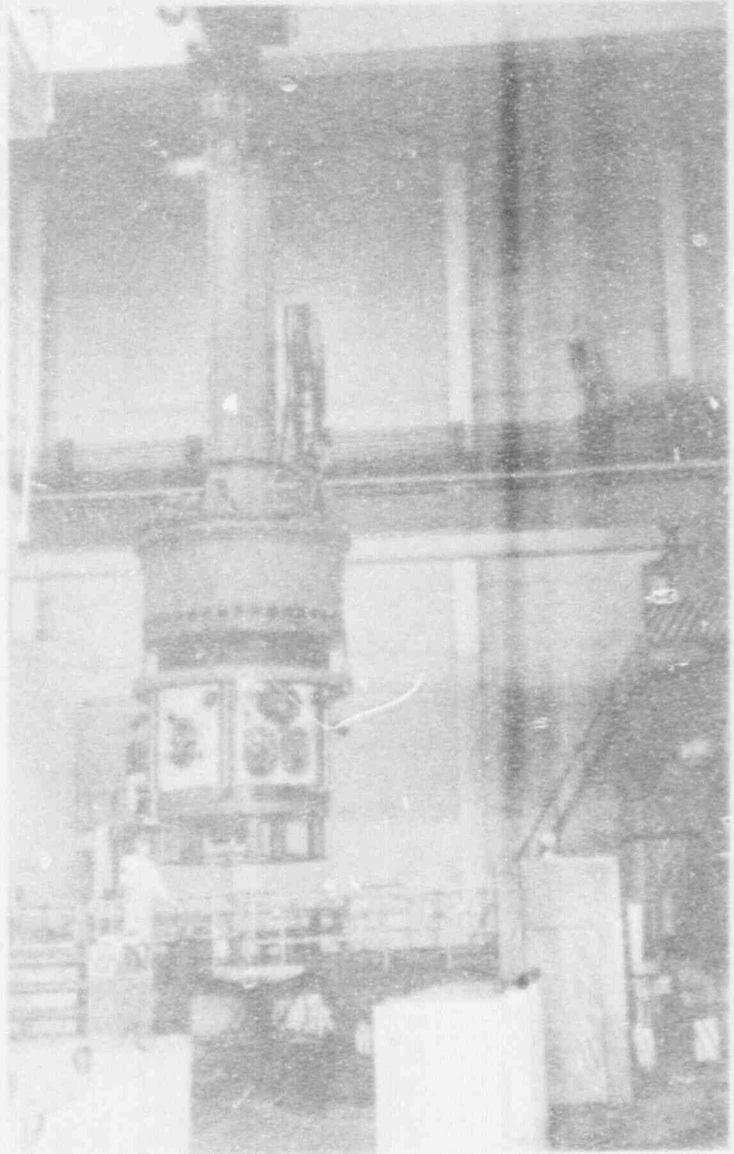
FIGURE 23

X. PHOTOGRAPHS



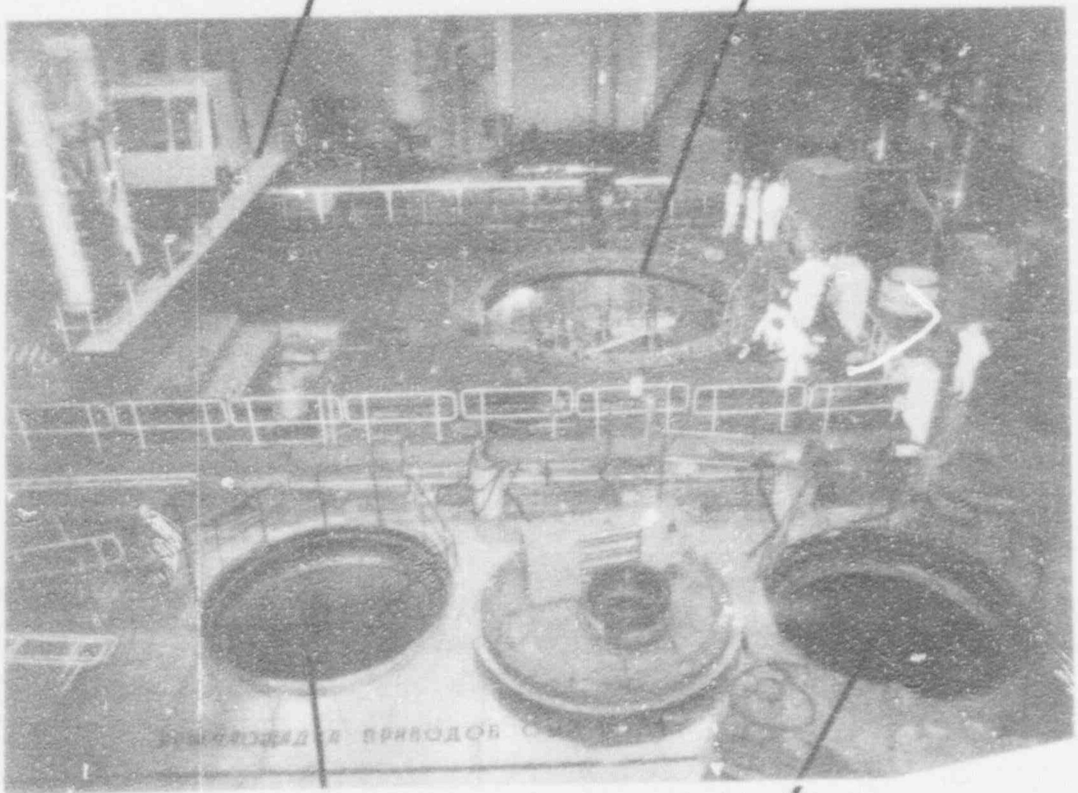
NOVOVORONEZH UNIT 3 AND UNIT 4
CENTRAL HALL OVERVIEW

PHOTOGRAPH NO.1



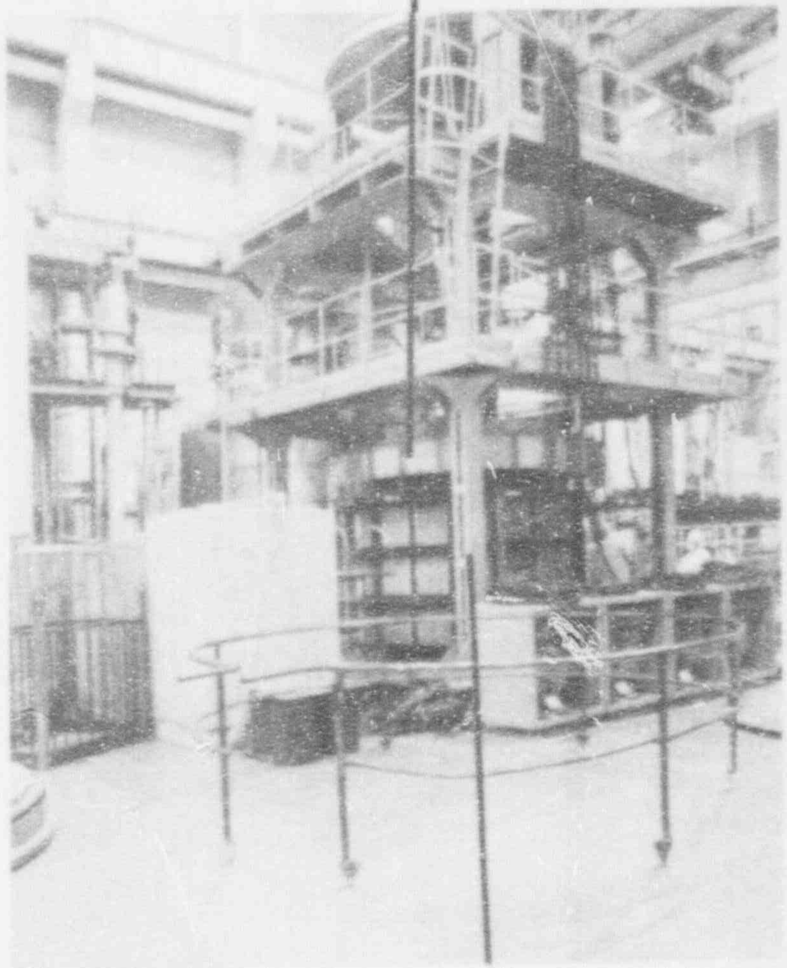
SHIELDED CABIN BEING REMOVED FROM
EMPTY REACTOR VESSEL

PHOTOGRAPH NO.2



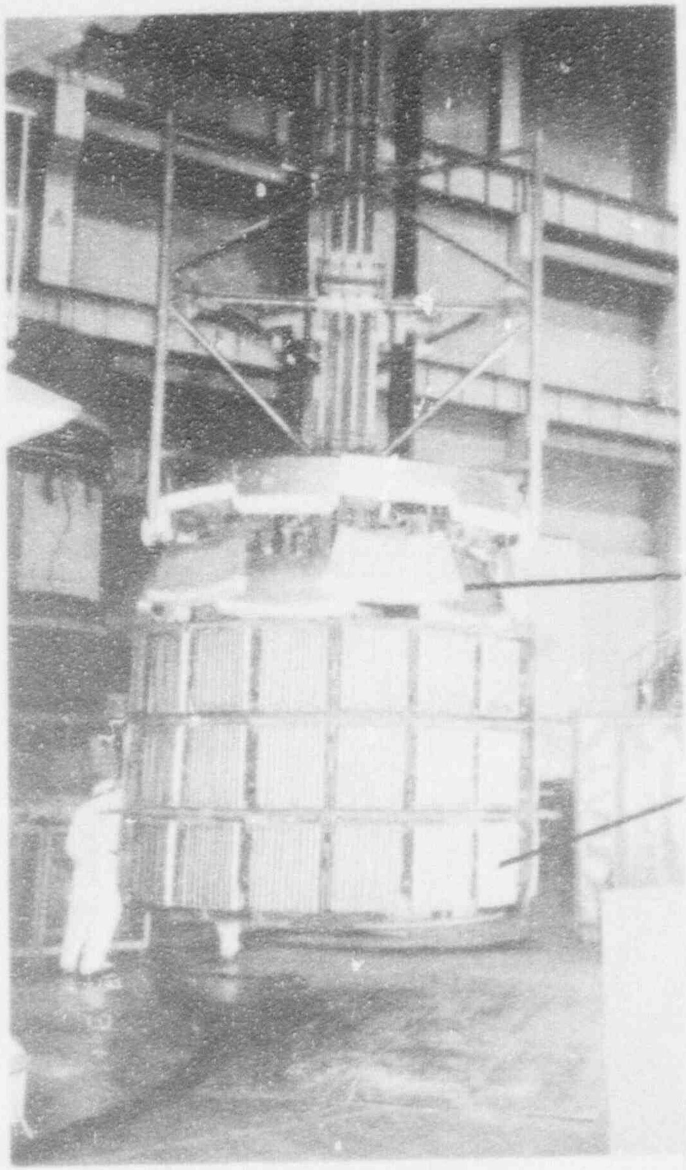
NOVOVORONEZH UNIT 3
REACTOR VESSEL REFUELING CAVITY

PHOTOGRAPH NO.3



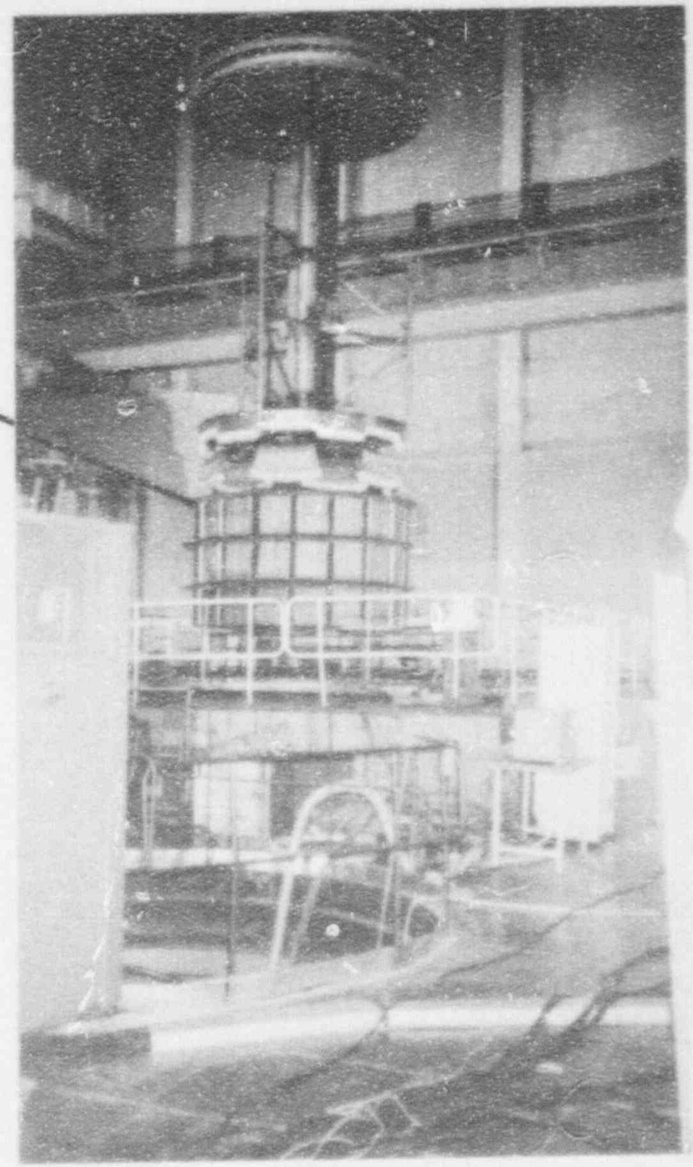
ANNEALING APPARATUS TEST STAND

PHOTOGRAPH NO.4



ANNEALING APPARATUS

PHOTOGRAPH NO.5



ANNEALING APPARATUS AS IT IS BEING LOWERED INTO THE REACTOR VESSEL

PHOTOGRAPH NO.6

U.S. TEAMS VISITING NOVOVORONEZH UNIT 3 FOR THE REACTOR VESSEL ANNEALING

I. Members of First U.S. Team and Soviet Contacts, February 2-24, 1991

- Kurchatov Institute Moscow, USSR, February 2 and 22, 1991
- Novovoronezh Power Station, Novovoronezh, USSR, Feb.3 - 21,1991

U.S. Participants:

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Keith R. Wichman - NRR	Thomas Friderichs
	Alex Zarechnak

Utilities:

William Beckius, Consumers Power
Stephen Collard, Florida Power and Light
Stephen Fournier, Yankee Atomic Electric Company
Stephen Chestnut, Pacific Gas and Electric
Michael Sullivan, Pacific Gas and Electric

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Kryukov, Alexander M.	Denisov, Sergei I.
Sokolov, Michael *	Kulakov, Gennady A.
Levit, Vladimir *	Luchkin, V.G.
Zaritsky, Sergei M. *	Sokolov, Fyodor
Proshina, Olga A. *	Burdin, Ivan
Khudoyarev, Alexander V.*	Frolov, S.V.
	Shatskich, Tatiana P.
	Pukhonin, Victor

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* Attended meetings at Kurchatov Institute only.

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Rogov, Michael F.	Zvezdin, Uri I. *
Dragunov, Y.G.	Vishkarev, Oleg M. *
Nosovsky, Alexander A.	

VNIIAES	Global (Contractor for EDM Sample Cutting)
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Kirilov, Valeri B. *	
Taborko, Irene N.	
Romanyk, V.I.	

Atomenergoproekt

Komarev, Gennady P. *

Ministry of Atomic Power and Industry

Lelekhin, Alexander O. *

Translator	Novovoronezh Training Center
Kavkovski, Victor I.	Lvov, N.K.

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* Attended meetings at Kurchatov Institute only.

II. Members of Second U.S. Team and Soviet Contacts, March 3-8, 1991

- Kurchatov Institute Moscow, USSR, March 1 and 8, 1991
- Novovoronezh Power Station, Novovoronezh, USSR, March 2 thru 7, 1991

U.S. Participants:

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Harold Gray - Region I	

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Alex Panagos - Commonwealth Edison

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Platonov, Pavel *	Khlustov, Alexander S.
Amaev, Amir D. *	Kulakov, Gennady A.
Kryukov, Alexander M. *	Verlin, Tom
Sokolov, Michael	Polyakova, Olga A. **
Shoob, Maxim L. **	

VNIIAES	Hydropress - Design Bureau
Romanyk, Viachelsav	Nosovsky, Alexander A.

Global (Contractor for EDM Sample Cutting)

Veselovsky, Oleg I.

---

* Attended meetings at Kurchatov Institute only.
** Translator

APPENDIX B

Novovoronezh Procedure

"MEASURES FOR KEEPING LIQUIDS FROM THE REACTOR VESSEL V-4M OF THE THIRD UNIT DURING REANNEALING DURING THE PLANNED PREVENTIVE MAINTENANCE (PPM - 90)

To prevent liquids from getting onto the internal or external surfaces of the reactor vessel during the annealing, the following must be accomplished 24 hours prior to installing the annealing rig inside the reactor:

1. Open the breakers for the pumps:

   H-1-6: (primary coolant pumps)
   H-1-3: (main feedwater pumps)
   H-1-6: (emergency feedwater pumps)
   and tag the associated controls:
   "Do not open - danger"

2. Coolant Loops 1-6 are to be drained fully, and the drains and vents are to be left open. The loop isolation valves are to be closed, the electrical feeds to the valve actuators are to be disconnected and the actuators locked closed. Removal of the actuators and reducers is allowed for conducting maintenance with the isolation valves closed.

3. All components associated with the reactor vessel, fuel pool, coolant loops and the pressurizer are to be placed in the condition indicated in Table 1.

4. Drain the quench tank fully and prevent its refilling.

5. After completion of Steps 1-4, prepare a report and enter the results in the appropriate logs.

During the annealing and cooldown process the following is required:

- During each shift confirm that all components are in their required state as defined in Table 1; enter the results of this check in the log.
- Visually check both dams between the reactor vessel and the spent fuel pool at least twice each shift.
- The level in the spent fuel pool is to be kept below 9.5 m as displayed in the control room as well as locally, at least twice each shift.
- All steam generators whose channel heads have been opened from the secondary side and those whose channel heads are scheduled to be opened on the primary side must be drained on the secondary side; the possibility of their refilling must be prevented.
- Decontamination is prohibited for the reactor vessel, the areas around the reactor, the pressurizer, the fuel pool and on the supports.
- Do not allow the flushing of the measuring and test instrumentation of the primary side, and of the reactor.
- The system for pumping the water from the reactor pit must be ready for immediate use. Checking for the absence of water in the sump of the reactor pit should be continuous and at least twice each shift visually.
- Testing of the sprinkler system with direct injection of water into the steam generator compartment and primary coolant pump compartments is prohibited.

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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(See instructions on the reverse)

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3. DATE REPORT PUBLISHED

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Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

A U.S. delegation attended the thermal annealing operation of the Novovoronezh Unit 3 reactor vessel in the USSR to evaluate the Soviet reactor vessel annealing technology and to determine its applicability to PWR reactors in the U.S. Operations observed and described in this report include reactor vessel sample cutting, preparations for annealing, installation of annealing apparatus, and initial heatup of the reactor vessel. The annealing operation witnessed has been developed to a routine operation and appears applicable to U.S. PWRs. Key areas requiring further work to confirm applicability to U.S. reactors are discussed.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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