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CHAPTER 11

THERMAL ANNEALING OF AN EMBRITTLED REACTOR PRESSURE VESSEL

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1. INTRODUCTION

Radiation embrittlement of ferritic pressure vessel steels i.e. usually manifested by a shift (increase) in the ductile-brittle transition temperature (DBTT), a drop in upper shelf energy as measured in the conventional Charpy V-notch test, and a decrease in fracture toughness (K_{Ic}, J_{Ic}). A thermal anneal cycle above the normal operating temperature of the vessel can restore most of shift in the ductile-brittle transition temperature, the original Charpy V-notch energy properties and the fracture toughness. While the possibility of restoring material toughness levels through thermal annealing treatments of reactor pressure vessels received little attention, it was not without precedent. Because of sensitive material and a low operating temperature of 239°C (430°F), the U.S. Army SM-1A reactor reached a point where thermal annealing of the reactor pressure vessel[(1)] was desirable after only a few years of operation. Interest increased in the late nineteen-seventies when the Electric Power Research Institute (EPRI) sponsored a programme to assess the feasibility of and to develop a methodology for thermal annealing an embrittled reactor pressure vessel[(2)]. Under the sponsorship of EPRI, Westinghouse clearly demonstrated the benefits of annealing in terms of mechanical property recovery[(2)]. In 1984, the BR-3 reactor vessel in Mol, Belgium underwent a "wet" anneal[(3)] similar to the "wet anneal" of the SM-1A reactor vessel[(1)]. About 15 commercial power reactor pressure vessels of the WWER-440 type have been successfully "dry annealed."[(3)] (See Section 4 of this chapter.)

2. THERMAL ANNEALING RESPONSE OF MATERIALS

The irradiation sensitivity and thermal annealing behaviour of reactor pressure vessels materials has been assessed by many investigators.[(4-41)] Spitznagel et al[(4)] studied the post irradiation annealing response of A302 grade B plate and high copper weld metal irradiated at 288°C and annealed in the temperature range 316°C to 454°C. Recovery of pre-irradiation microhardness was followed for all three steels. The measured annealing response of these steels agreed well with the values predicted by a previously developed theoretical model based on the dissolution of copper-vacancy aggregates. A number of authors presented the results of their studies at the 1979 IAEA specialist meeting in Vienna, Austria. Steele[(5)] presented a review of the IAEA specialist meeting. Sivaramakrishnan[(6)] carried out post irradiation annealing studies on the two types of steels utilized in the TAPS and RAPP reactors in India and reported that post irradiation thermal treatment can recover irradiation damage. Hawthorne[(7)] reported on post irradiation heat treatment (annealing) as a method for reversal of detrimental radiation effects for reactor vessel steels. Petrequin and Soulat[(8)] reported on thermal annealing studies carried out in France. Annealing of irradiation embrittlement was examined on a 508 class 2 reactor pressure steel by heat treatment in the range 300° to 500°C. As with other published data recovery was a function of annealing temperature. Popp et al[(9)] presented a set of experimental data regarding the influence of sampling depth, neutron irradiation and post-irradiation annealing treatment on properties derived from instrumented Charpy V-notch impact testing, tensile and hardness tests. Complete recovery was reported for Charpy impact energy transition temperatures, hardness, and dynamic yield stress. The most effective recovery took place during short-time anneals of about one hour.

In a programme sponsored by the United States Nuclear Regulatory Commission (NRC) and conducted by Hawthorne et al[36,37,38] post irradiation annealing and re-irradiation behaviour of plate and weld were investigated. The influence of factors such as copper, nickel and phosphorus contents, weld flux and annealing temperature (399 and 454°C) were studied. Phosphorus had no influence on annealing, but copper and copper plus nickel reduced the annealing affect.

In Germany, post irradiation annealing and re-irradiation experiments on a NiCrMo type weld containing 0.27% Cu (on average) were conducted in the nineteen seventies but published later.[39,40,41] Hardness and Charpy-V tests were conducted after annealing at several temperatures, drop weight and fracture mechanics tests after annealing at 450°C. They showed that annealing at 450°C for 60 hours produced significant irradiation recovery.

A number of papers reported on the development of a model of annealing and reembrittlement consistent with available experimental data that can be applied to commercial reactor pressure vessels. A model based on copper precipitation was reported by Lott, et al[10] which concluded that the model underpredicted the reembrittlement rate for weldments. Originally, Lott et al[10] considered two possible mechanisms for annealing; dissolution of copper precipitates and over-aging by an Ostwald Ripening process. The copper concentrations in reactor pressure vessel steels appear to be well in excess of the solubility limits in the temperature range 300°C to 400°C. Therefore, dissolution appeared to be the least likely mechanism. In Ostwald ripening, the larger precipitates grow at the expense of the smaller ones. Pachur[11] identified four distinct annealing stages. Pachur's work has several important implications for the interpretation of the power reactor annealing data. First, the existence of two and possibly three mechanisms of embrittlement indicates that a simple single mechanism model based on copper precipitation will not adequately describe annealing. The existence of a mechanism that is marginally unstable at 290°C helps to explain why the damage produced at lower irradiation temperatures is more easily recovered than damage produced at reactor pressure vessel operating temperatures. Extrapolations based on experience with lower temperature anneals may produce erroneous conclusions. Pachur also indicates that the drop in Charpy upper shelf is caused by type 3 defects and not by type 4 defects. This would explain the results of a previous annealing study, where it was found that the Charpy upper shelf was completely recovered after a 450°C anneal while the Charpy transition temperature was not. Earlier, Nichols[12] published a model for the recovery (annealing) of radiation damage in pressure vessel materials and suggested that annealing of the damage occurs by the thermally activated motion and coalescence of the damage zones as entities through surface diffusion of atoms on the zone. Fisher and Buswell[13] extended and adapted the "Magnox model"[14] to the interpretation of data from PWR pressure vessel steels. The extended Magnox model of Fisher and Buswell[13] can be very helpful in assessing the response of irradiated thermal treatment (annealing). The extended Magnox model suggests that as the irradiation temperature decreases the contribution of displacement damage increases and as the irradiation temperature increases the contribution of copper precipitation increases. More recently, Macdonald[15] developed a correlation to describe the residual shift in the 41 joule Charpy transition temperature after thermal annealing of reactor pressure vessel steels. Macdonald[15] expanded on the previous work of Powers[16] and resulted in a correlation reasonably current for residual post-annealing shifts after annealing of less than 111°C.

The annealing and re-irradiation response of typical reactor pressure vessels is given in Figures 1, 2, and 3.[2,17] Conclusions that can be drawn from references 2 and 17 are:

- The ductile-brittle transition temperature shift recovery is between 80 and 100% after annealing at 454°C for 168 hours.

- The Charpy upper-shelf impact energy recovery is a 100% after annealing at 454°C.

- The re-irradiation after annealing is a function of the annealing temperature. At high annealing temperature (454°C) the ductile-brittle transition temperature appears to continue at the lower rate which would have been expected if no anneal had been performed. At lower temperatures (<400°C) the ductile-brittle transition temperature appears to have a lateral shift[17].

The results of Leitz et al[39,40,41] confirm the above statements. In addition, they found that hardness also (as upper shelf energy) recovers completely at 450°C for 60 or 168 hours. Transition temperatures, even after 500°C annealing, are not completely recovered but after a 550°C anneal the total transition curve is an improvement on that before annealing.

A. D. Amaev, et al[18] reported test results of specimens removed from operating VVER-type reactors. Analysis of 40 sets of vessel material specimens were conducted to assess the role of copper and phosphorus. The results of the investigation of annealing effect on recovering the brittle transition temperature included those of material properties after the repeated cycle of radiation following annealing. Figure 4 shows the change of the transition temperature of the vessel materials depending on the neutron fluence with periodic phases of radiation and annealing at 420°C for 144 hours. It is seen in Figure 4 that the degree of the steel radiation embrittlement after annealing did not depend on the material history and was only determined by the condition of the final anneal.

3. THERMAL ANNEAL PROCEDURE DEVELOPMENT

3.1 U.S. Developments

There are two methods of thermal annealing an embrittled reactor pressure vessel: wet and dry. The U.S. Army SM-1A and BR-3 reactor vessel used "wet" annealing for their thermal annealing programme. "Wet" annealing restricts the temperature of the thermal anneal to the design temperature of the nuclear steam supply system (NSSS). At this relative low temperature, full recovery of the mechanical properties is not possible and upon re-irradiation after annealing the increase in the DBTT shift will continue at a similar rate which would have been expected if no anneal had been performed. "Dry" annealing permits selection of an elevated temperature to obtain full recovery of mechanical properties. Therefore "dry" annealing is recommended and a thermal anneal procedure was developed to utilize "dry" annealing.

To assess the feasibility of and to develop a procedure for dry thermal annealing, the following objectives must be met:

- determine the thermal annealing temperature and holding time required to maximize fracture toughness recovery, minimize re-exposure sensitivity, and minimize reactor downtime.

- determine the areas of the reactor vessel that may require thermal annealing.

- determine if the following restrictions would hamper the feasibility of applying a thermal annealing operation:

    - reactor vessel design and metallurgical limitations
    - reactor vessel insulation limitations
    - primary shield concrete limitations
    - RCS piping and associated equipment support restrictions
    - reactor vessel internals and fuel storage limitations
    - health physics considerations

- determine the most efficient method of producing and maintaining the thermal annealing parameters. Possible methods are as follows:

    - spent fuel assemblies used as heating elements
    - circulating externally (outside vessel) heated fluid (inert gas)
    - induction heating elements
    - circulating internally (inside vessel) heated fluid (hot air)
    - resistance heating elements

- develop a conceptual apparatus to be used to thermally anneal a reactor vessel.

- develop an in-situ thermal annealing procedure.

- develop technical specifications related to the thermal annealing procedure.

In Germany, as a provision, annealing equipment for annealing the central circumferential weld seam at the beltline of a reactor pressure vessel at 450°C was designed, constructed, and tested on a dummy vessel[40]. Later on it was shown, however, that the annealing procedure was not needed.

The results of the EPRI work [2] indicated that a thermal annealing method based on dry, localized heating could be used to apply the annealing temperature of 454°C for 168 h in an effort to restore the fracture toughness properties of ferritic materials of the reactor vessel as required by 10CFR Part 50 Appendix G[19]. The method would require an apparatus designed so that it could be lowered on the reactor vessel. After a watertight seal is formed between the inside of the reactor vessel and the refueling cavity, the primary water inside the reactor vessel would then be pumped out into the cavity. The resistance type heating elements system, designed in a manner so that the area requiring thermal anneal would be heated, could then be cycled on and off to achieve the specified thermal annealing parameters. This method could be adapted to the standard two-, three-, and four-loop plants. Removal of the reactor vessel internals and the fuel assemblies would be required. The maximum volume of a reactor vessel that may require a thermal annealing treatment would comprise that material from some point below the beltline region up to and including the intermediate-to-upper shell girth seam. This volume is described in Figure 5.

3.2 Thermal Annealing Equipment

In developing the EPRI conceptual thermal annealing equipment, the following factors were considered:

- ability to produce and maintain a temperature of 454°C for a holding time of 168 hours.
- ability to control the heatup rate to less than 28°C/hour.
- power requirements
- costs
- logistics

The annealing apparatus (Figure 6) would be lowered on the reactor vessel with the assistance of the overhead polar crane. A watertight seal would be made between the support plate (4) and the reactor vessel internals support ledge (18) to prevent leakage between the inner confines of the vessel (24) and the refueling cavity (16). This would be a secondary water seal. The primary water seal would be made between the water shield barrel dam (8) and the bottom of the refueling cavity (16) by the use of a new design cavity seal ring (5).

3.3 Generic Thermal Annealing

The following would be the general thermal annealing procedure on a generic basis.

(1) All unassembled thermal annealing equipment would be moved into the containment building through the equipment hatch. This equipment would then be transferred to the operating deck.

(2) The missile shield, cavity gates, seismic restraints, control rod drive mechanism, vent air ducts, and closure head areas insulation would be removed and stored per the standard operations procedure.

(3) The reactor vessel studs would be detensioned, unthreaded, and then removed per the standard operations procedure.

(4) The new design cavity seal ring for adapting the thermal annealing water shield barrel dam would be installed.

(5) The closure head would be removed and the refueling cavity flooded. Then the upper internals would be lifted and placed in storage in the reactor vessel internals storage area. In parallel with this procedure, the thermal annealing equipment would be prepared for assembly.

(6) All fuel assemblies would be removed and transferred to the spent fuel pit. In parallel with this procedure, the thermal annealing equipment would be assembled.

(7) After all fuel assemblies have been transferred, the lower internals would be lifted and placed in storage in the reactor vessel internals storage area.

(8) The annealing water shield barrel dam would be lowered by the overhead polar crane and sealed to the floor of the refueling cavity.

(9) The reactor vessel would be drained to just below the vessel flange. The water would be pumped out of the inner shield barrel dam volume into the remainder of the refueling cavity.

(10) The seals on the water shield barrel dam would be checked for leaks and a segment of the cavity seal ring would be removed thereby operating the air path between the lower reactor vessel cavity and upper containment.

(11) The reactor vessel would be drained to a water level just below the reactor vessel nozzles.

(12) The nozzle plugs would be installed.

(13) The thermal annealing apparatus package would be lowered onto the reactor vessel by the overhead polar crane. All vent and drain lines would be checked.

(14) The remaining water would be pumped out of the vessel into the refueling cavity.

(15) The resistance heaters would be energized using a low voltage power source to dry out the heating elements. All electrical connections, heater controllers, X-Y recorders, and thermocouple functions would be checked.

(16) The resistance heaters would be cycled on and off using a high voltage power source to achieve and maintain the annealing temperature (454°C) for the required period of time (168 hours).

(17) The vessel would be permitted to cool to ambient temperature.

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(18) The cavity seal ring segment, removed previously, would be installed.

(19) Water would be transferred from the refueling cavity to the reactor vessel, to a level just below the nozzles.

(20) The thermal annealing apparatus package would be removed from the vessel by the polar crane, and the package would be decontaminated.

(21) The nozzle plugs would be removed and decontaminated.

(22) The inner confines of the water shield barrel dam would be filled with water to equalize the water levels with the refueling cavity.

(23) The water shield barrel dam would be removed, decontaminated, and disassembled.

(24) All equipment would be packaged and removed from the containment building.

(25) The lower internals, all fuel assemblies, and the upper internals would be installed to standard operations procedure.

(26) After draining of the refuelling cavity, the closure head would be installed to the standard operations procedure.

(27) The cavity seal ring would be removed.

(28) The reactor vessel studs would be installed and tensioned to the standard operations procedure.

(29) The closure head insulation, control rod drive mechanism vent air duct, seismic restraints, cavity gates, and missile shield would be installed.

4. ANNEALING OF WWER PRESSURE VESSELS

Background studies[20] on the heat treatment of WWER-440 reactor vessels and the development of processes and hardware by the leading organizations in the then USSR led to the pilot operation annealing of Novo-Voronezhskaya NPS, Unit III by "dry" method (430 ± 20°C, 150 hours). In 1988 the Armenian NPS, Unit I, was annealed by a revised procedure (450 + 40°C, 150 hours). The experience of performing these two first recovery heat treatments of operating power reactors verified the procedures adopted. The annealing of eight more WWER-440 vessels of reactors has been carried out: NPS "Bruno Leuschner," Unit I, GDR (1988); Kolskaya NPS, Units I and II, USSR (1989), NPS Kozlodui; Units I, II and III, Bulgaria (1989 and 1992), NPS "Bruno Leuschner," Units II and III, GDR (1990).

The investigations showed that for the assurance of safe operation and life extension of radiation of WWER-440 reactor vessel with regard to the brittle failure strength criterion would be met by annealing the weld in the high fluence area of the pressure vessel which was the controlling region with regard to irradiation embrittlement. This feature eliminated many technical difficulties if annealing of the entire reactor vessel had been needed and made the process of restoration of properties of vessel metal practicable.

The heating equipment was developed and fabricated by NPO "TZNITMASH" and PO "Izhorskyj zavod." The heating equipment (Figure 7) is a welded construction incorporating a heating unit, a cover, being both a biological and thermal protection, and a cross-piece connecting the heating unit and the cover. The heating unit has 54 panels of one-sided heaters of radiative heaters arranged into three circular rows, each row has 18 electrical heaters. Each row of electrical heaters is divided into 3 independently controlled areas having two paired thermoelectric temperature transducers (thermocouples) to monitor the vessel temperature in the area of annealing and the total number of thermocouples is eighteen.

The heating equipment also includes a distributing board, power cabinets, and transformers (one for each heating zone). The transformer power is 90 Kv. Also included are cabinets to control the heaters; a system of power supply communications, monitoring and control circuits, a system for automatic maintaining the annealing conditions by the assigned programme (Remicont), a system for recording thermocouple readings during the anneal, a system for remote control and monitoring thermal and electrical conditions, a jig for assembling and adjusting and temporary storage of the equipment.

Before the start of vessel annealing the following procedures are followed, emptying the reactor vessel, its drainage, cleaning and drying, checking the equipment cooling system of the reactor pit to prevent overheating of structure surrounding the vessel during annealing. Also measures are taken to exclude water, cold air and other materials into the reactor vessel during heat-up, hold-time during annealing and cooling down:

- work on steam generators, reactor coolant pumps (RCP), pressurizer is forbidden;
- closure of all main gate valves (MGV) is checked;
- work on non-disconnected sections of the main coolant circuit is prohibited;
- electrical circuits of the RCP and MGV are dismantled;
- the hydroseal of the cooling pond and refueling channel are leak tested;
- closing valves of pulse tubes of inspection of the reactor main joint leakages is checked;

- closure of the manhole in the thermal insulation of the vessel in the bottom area is checked.

After erecting the jig, assembling the heating equipment, connection of electrical equipment and unit-by-unit inspection of thyristor controls at the nominal current load, the heating apparatus is put into the reactor vessel and located on its main joint. The equipment design provides location at the middle ring of heaters opposite the axis of the weld being annealed (the annealing zone covered by the heating equipment is not less than ± 0,7 m with respect to the weld axis).

The following temperature-time conditions were used to anneal the WWER-440 welds:

- annealing temperature of 475 ± 15°C;
- holding time of 150 hours.

Annealing conditions are monitored with thermocouples. After the equipment has been inserted into the reactor vessel, a special mechanism moves the thermocouples and presses them to the internal surface of the vessel wall. The same mechanism presses the thermocouples tight against the vessel wall during its expansion during annealing. Additional thermocouples are installed outside the reactor vessel (nozzle area, supporting shoulder, cylindrical part and bottom), in the channels of ionization chambers of the tank of biological shielding, on the concrete of the reactor pit and on the thermal insulation of the vessel (Fig. 8). The number of additional thermocouples is not less than twenty. Thermocouple readings are continuously recorded.

The following restrictions apply during annealing:

- the heating up rate is not higher than 20°C/h;
- the cooling down rate is not higher than 30°C/h;
- the temperature of the vessel supporting shoulder is not to exceed 300°C;
- the temperature of the biological shielding tank water is not to exceed 90°C;
- concrete temperature of the reactor pit in the area of tank fastening – not higher than 90°C;
- stratification of temperature by the readings of thermocouples of one positions is not to exceed 50°C.

During annealing routine calculations of the thermal state of the vessel and pit equipment components are carried out. Temperature trends are analyzed and evaluated and any corrections needed are made.

Annealing conditions are at temperatures above the operating temperature (annealing temperature is 490°C). Heat flows from the region being annealed to the top and bottom regions of the vessel. Because of this, a major fraction of energy which the vessel receives from heaters is spent in heating the vessel. Some fraction of energy is lost through the thermal insulation and there is a heat sink through the supporting structure.

Local heating of the annealing area produces stresses and these result in deformation of the reactor vessel. Maximum stresses act at the edge of the annealing area at when the highest temperature gradients occur both along the height and through the thickness of the vessel wall. At the hold stage as the vessel is being warmed up, stresses decrease. Stresses arising in the vessel during annealing should not be at a level to cause metal creep, residual strains and cause stresses which could prevent reactor vessel safe operation. Implementation of annealing conditions must not result in overheating of structures surrounding the vessel. Mathematical models and corresponding computer programmes were developed to analyze the thermal-stress state of the reactor vessel during annealing.

In all actual vessel anneal cases thermal conditions of the vessel and reactor pit equipment as a whole corresponded to the design one and temperature of separate components of the vessel and pit equipment did not exceed that permitted. Check calculations of vessels strength performed using temperatures actually measured during annealing verified the fact that stresses did not exceed the design values.

Prior to and subsequent to annealing the vessel was inspected nondestructively. The results show that annealing does not initiate new, or indeed develop flaws.

When the hold stage (150 hours at a temperature of 475$\pm$ 15°C) is completed then the heating equipment is turned off and the vessel is gradually cooled down to temperature not more than 70°C and then the vessel is filled with water.

The evaluation of the vessel metal properties and quality control of the annealing process are matters of great importance. This activity encompasses the determination of variation of mechanical properties, and the investigation of the degree of recovery of mechanical properties by means of metal sampling and also the measurement of hardness (for vessels without cladding).

The VNIIAES Institute has developed and fabricated equipment for sampling weld metal from the reactor vessel inner surface in the form of a 'chip' to conduct chemical analysis from the first 5 mm of the vessel wall and then grind the locations of sampling. The equipment includes a milling cutter device, a chip collector, grinding device, control cabinet, and a rig for simulating and adjustment of facilities and devices.

To perform the work the milling cutter and grinding devices are hung on the multipurpose maintenance cabin (MMC) or on the protective container (PC). Metal sampling is carried out along the weld axis in three places at an angle 120° to one another.

Chemical analysis of the sampled chip allows the determination of phosphorus and copper in the weld metal and to check the brittle strength of the heat treated reactor vessel. Table 1 represents values of phosphorus and copper content for the reactor vessels subjected to annealing.

Hardness is measured prior to and subsequent to heat-treatment with the help of an automatic hardness meter T-4M (designed and fabricated by VNIIAES to determine the variation of mechanical properties of reactor vessel base and weld metal. The hardness meter is a multipurpose machine for establishing values of Brinell (HB) hardness of metal of the reactor vessel linear surface. The apparatus has a measuring head, a moveable carriage with magnetic fastening, a control desk, a unit for processing the results, and for operation it is hung onto the multipurpose maintenance cabin or protective cabin in a similar way to the metal sampling equipment.

The machine is fixed on the internal surface of the reactor vessel with a magnetic frame (pressure force 900-1200 kgf), and after that the moveable carriage with the hardness meter head attached on it is set with respect to the selected measurement point. The entire process of loading and load relief is carried out in an automatic way with simultaneous recording using a two-coordinate recorder and a minicomputer memory. The range of hardness measurements is 10 – 500 HB. The results are processed on the computer immediately after finishing the complete cycle of measurement.

Hardness measurements of weld and base metal of adjacent shells is carried out on non-ground surfaces at three points in close vicinity to the metal sampling positions above and below the weld axis. Measurements are made at each location before and after annealing for 10 loading cycles and with a distance between measurement points of 10 – 15 mm.

Hardness results (See Table 2) clearly demonstrate the decrease of hardness values after annealing and weld metal hardness is reduced to a greater extent..

After annealing the NPP Bruno Leuschner Unit I, and before and after annealing the NPP Bruno Leuschner Unit II and Kosloduy Unit II, Samples were removed from the inner surface of the RPV and these were big enough to provide subsize V-notch impact specimens. The results of been reported elsewhere (25) and they showed that the transition temperature after annealing was lower than predicted. (Testing of samples of Bruno Leuschner" was discontinued after the German Authorities decided not to continue with the operation of the plant. It is proposed to complete the tests as part of the general watch background for WWER 440/230 reactors). The Kosloduy II samples are being tested as part of a cooperative effort by Electricite de France, the Kurchatov Institute and Prometey Institute, and Siemens. The results are not yet available for publication.

The work schedule for reactor vessel annealing including metal sampling and measurements of hardness before and after heat treatment amounts to about 15 days which is included in the maintenance schedule of the unit.

At present there are two sets of equipment for annealing WWER-440 reactor vessels. This factor allows up to 4 annealings per year. The heating equipment is collapsible in design and consists of separate transportable sections permitting thorough decontamination and, fits into containers that can be transported to NPS by road, rail or water.

5. RECOMMENDED GUIDE FOR THERMAL ANNEALING

The American Society for Testing Materials (ASTM) has issued a recommended guide which covers the general procedures to be considered for conducting an inservice thermal anneal of a light-water-cooled nuclear reactor vessel and demonstrating the effectiveness of the procedure. The guide, ASTM E509, "Standard Recommended Guide for Inservice Annealing of Light-Water Cooled Nuclear Reactor Vessels,"[21] is designed to accommodate the variable response of reactor-vessel materials to post-irradiation heat treatment at various temperatures and different time periods. The guide describes certain inherent limiting factors which must be considered in developing an annealing procedure; these factors include (a) system design limitations, (b) physical constraints resulting from attached piping, support structures, and the primary system shielding, and (c) the mechanical and thermal stresses in the components and the system as a whole. The guide also provides guidance for developing a post-annealing surveillance vessel radiation surveillance programme to monitor the effects of subsequent irradiation of the annealed-vessel beltline materials.

Appendix B, "Guidance for Verifying Recovery and Re-Irradiation Embrittlement," to ASTM E509 (21) addresses the new rate of re-embrittlement after annealing. The guide introduces the concept of horizontal or lateral shift of the initial irradiation embrittlement path to become the post-anneal re-irradiation trend curve and vertical shift (vertically downward). The generally accepted concept or model for the rate of re-embrittlement is the lateral shift.

The U.S. Nuclear Regulatory Commission (NRC) is in the process of preparing a NUREG that will address reactor pressure vessel annealing. It is expected that NRC will incorporate the basis and concepts identified in ASTM E509 into the NUREG.

6. SUMMARY

The use of a thermal heat treatment to recover mechanical properties which were degraded by neutron radiation exposure is a potential method for assuring reactor pressure vessel compliance with regulatory licensing rules and for license renewal. Research programmes worldwide have clearly demonstrated the recovery effects on annealing irradiated steels and weldments. "Wet anneals" at temperatures less than 343°C have been conducted in both the United States and Belgium (SM-1A reactor and BR3 reactor). The Russians and other countries in eastern Europe have performed "dry anneals" at higher temperatures (450°C) than design temperatures (343°C) for a number of commercial reactor vessels.

In the United States, a standard recommended guide for inservice annealing of reactor pressure vessels has been published by ASTM[21]. This recommended guide is expected to serve as the basis for a NUREG to be issued by the US NRC on inservice annealing of reactor pressure vessels in the United States. Technical and economic uncertainties have made utilities in the United States reluctant to seriously consider thermal annealing to date of their reactor pressure vessels. However, as utilities begins to experience significant radiation embrittlement or consider extending the operating licensing life of the vessel, thermal annealing of the reactor pressure vessel will take place. Griesbach and Server[22] have shown that thermal annealing of an embrittled reactor pressure vessel is of economic benefit.

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TABLE 1
PHOSPHORUS AND COPPER CONTENT DETERMINED ON METAL SAMPLES OF WELD CONTENT BEING ANNEALED

NPS	Unit No.	Content of Elements in Metal of Weld No. 4, %							
		Phosphorus (P)				Copper (Cu)			
		Actual			Used in Design Verification	Actual			Used in Design Verification
		Min	Max	Avg		Min	Max	Avg	
NV NPS	3	0,0336	0,0401	0,038	0,039	0,123	0,138	0,130	0,21
A NPS	1	0,032	0,034	0,033	0,030	0,110	0,110	0,110	0,16
"Bruno Leuschner" NPS	1	0,032	0,034	0,033	0,036	0,086	0,104	0,092	0,12
"Bruno Leuschner" NPS	2	0,033	0,037	0,035	0,032	0,150	0,157	0,1535	0,18
Kolskaya NPS	1	0,0306	0,0331	0,032	0,037	0,121	0,146	0,134	0,21
Kolskaya NPS	2	0,0317	0,0375	0,0346	0,036	0,138	0,154	0,145	0,12
"Kozlodui" NPS	1	0,028	0,036	0,032	0,036	0,100	0,120	0,110	0,12

HARDNESS RESULTS FROM THE REACTOR VESSEL OF KOLSKAYA NPS, UNIT I, PRIOR AND SUBSEQUENT TO ANNEALING					
Place of Measurement		Hardness Values Prior to Annealing HB (kgf/sq.mm)		Hardness Values Subsequent to Annealing HB (kgf/sq.mm)	
		(HB (irrad.)	$HB_{aver.}$(irrad.)	HB (anneal.)	$HB_{aver.}$(anneal.)
1	Upper Shell	266 221	223,5	184 195	189,5
	Weld	243 247	245,0	198 199	198,5
	Lower Shell	224 219 217 229 232 233	225,6	184 193 187 185 193	188,4
2	Upper Shell	228 218 217	224,3	189 201 201	197,0
	Weld	240 243 241	241,3	180 210 207 200	199,3
	Lower Shell	213 230 230	224,3	191 193	192,0
3	Upper Shell	219 221 220	220,0	188 184 190	187,3
	Weld	236 231 242	236,3	203 201	202,0
	Lower Shell	232 225 223	226,6	198 186 190	198,0

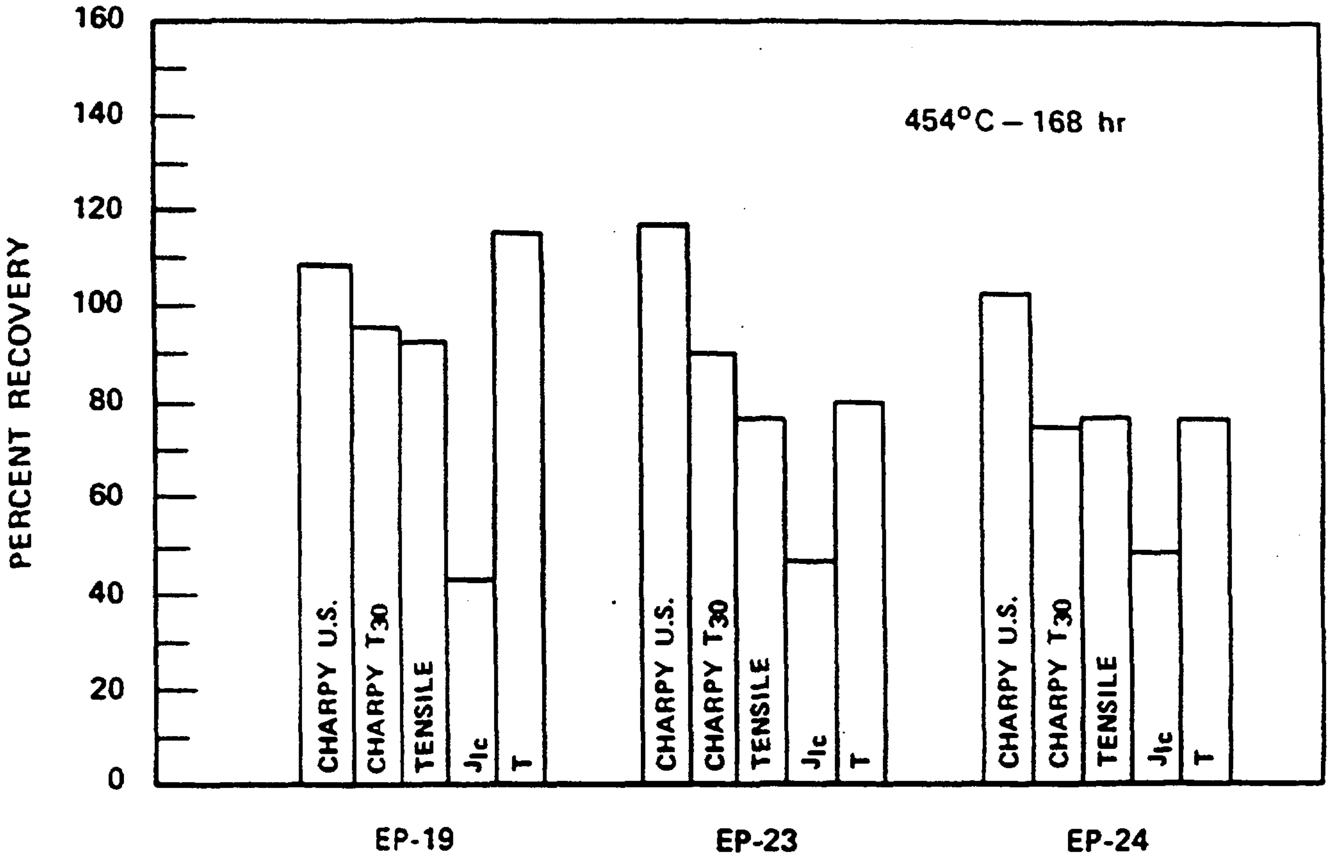


Figure 1. The Ductile-Brittle Transition Temperature Shift Recovery Between 80 and 100% After Annealing at 454°C for 168 Hours.

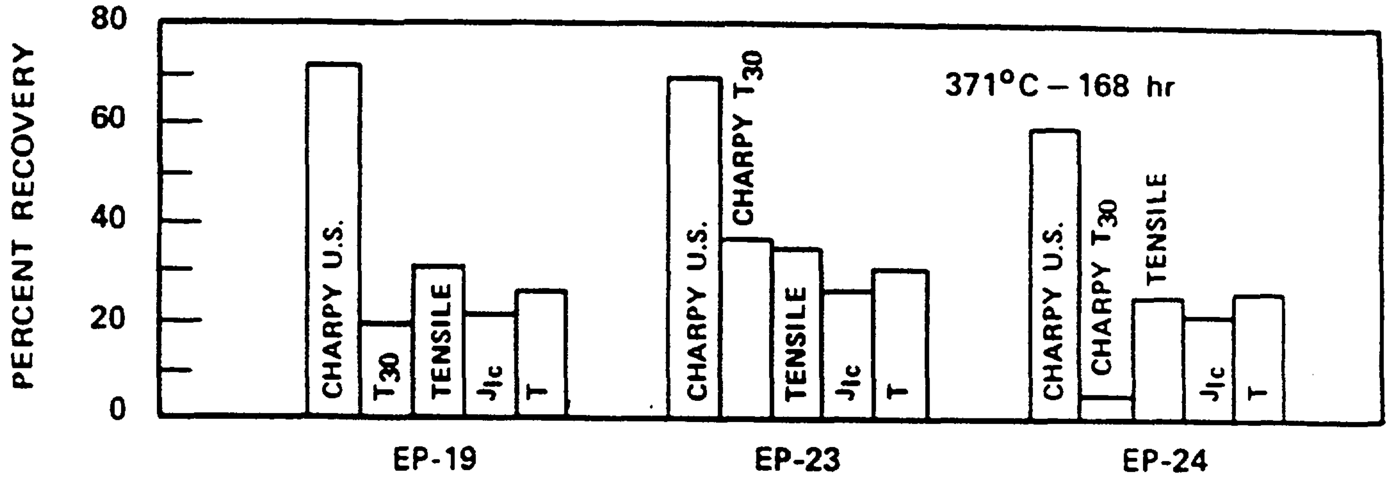


Figure 2. The Ductile-Brittle Transition Temperature Shift Recovery Between 20 and 40% After Annealing at 371°C for 168 Hours.

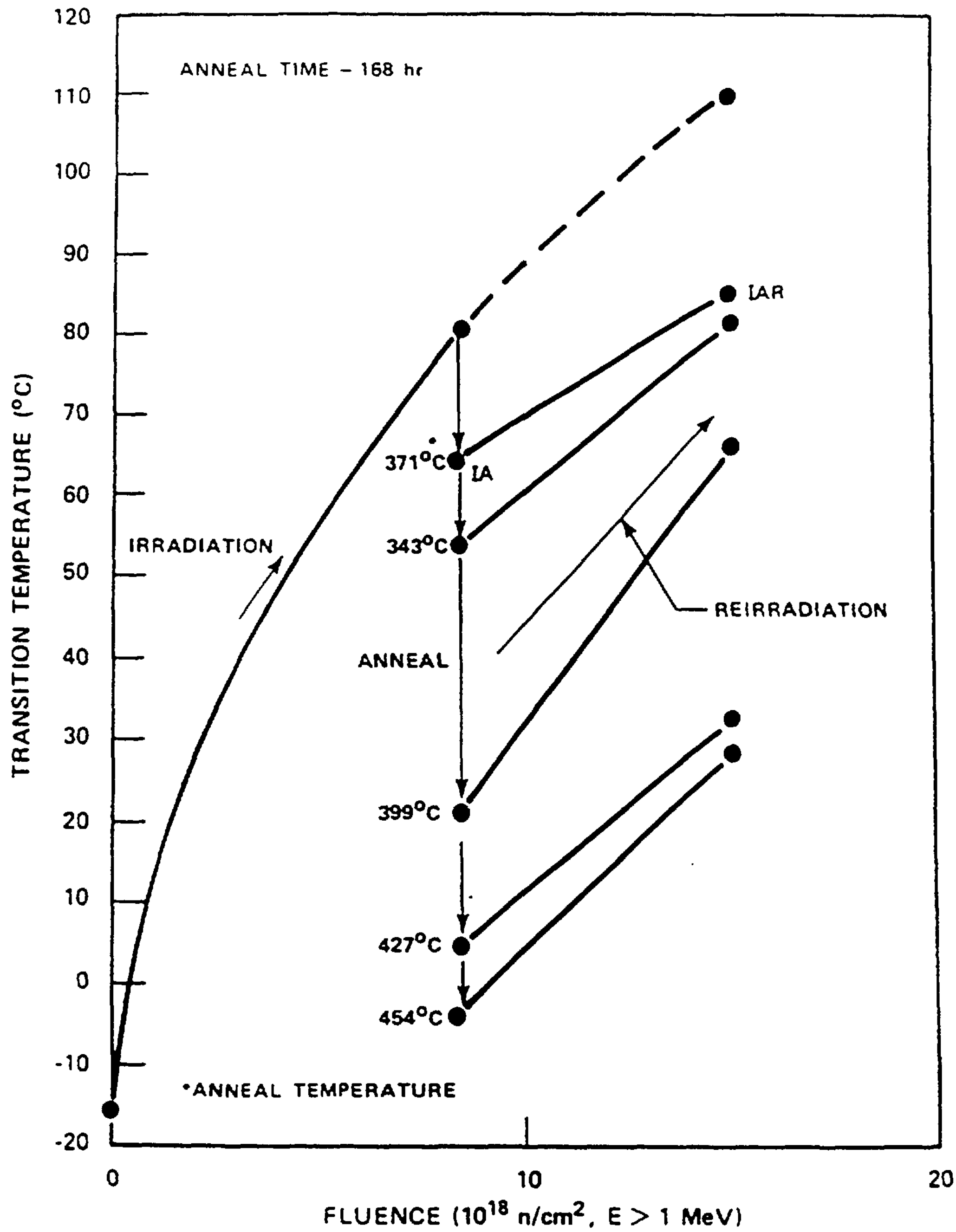


Figure 3. Re-Irradiation After Annealing is a Function of the Annealing Temperature.

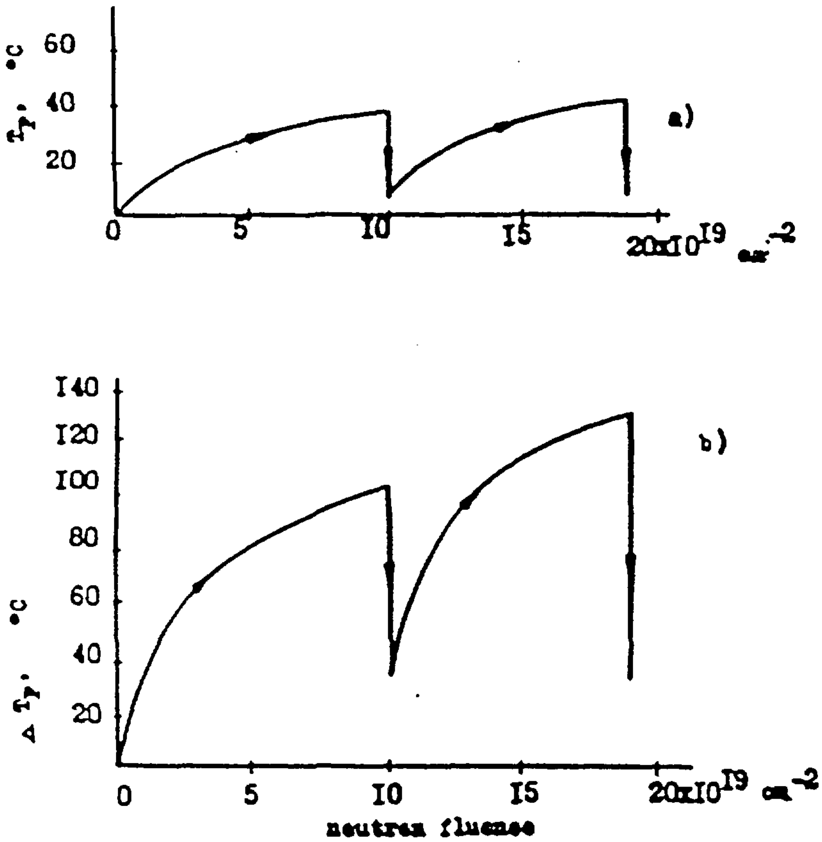
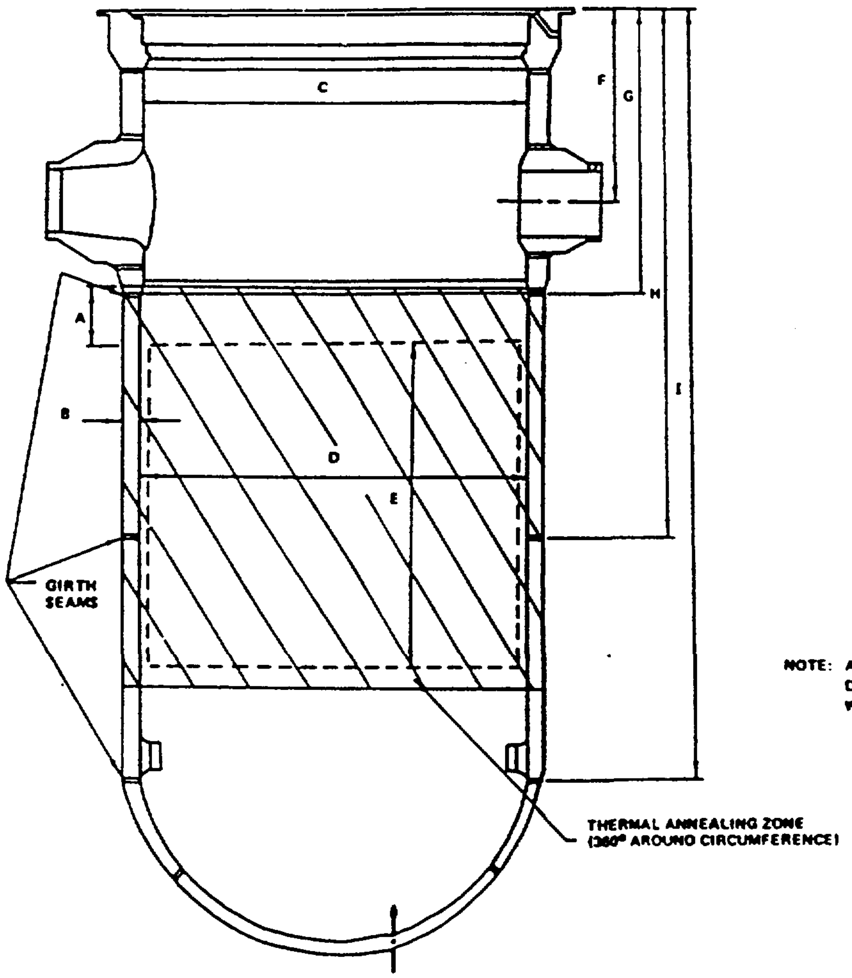


Figure 4. Diagram of Change of Critical Brittle Temperature of 15x2 MФA Steel (a) and its Weld Metal (b) Depending on the Neutron Fluence at Periodical Alternation of Irradiation and Annealing at 420°C (During 144 Hours).



LETTERS	DIMENSIONS (in)		
	CASE 1[a]	CASE 2[b]	CASE 3[c]
A	25	25	25
B	6.6	7.9	8.6
C	132	156	171
D	132	157	173
E	144	144	144
F	75	84	85
G	117	128	126
H	220	233	233
I	324	338	340

a. CASE 1 – TWO-LOOP REACTOR VESSEL
b. CASE 2 – THREE-LOOP REACTOR VESSEL
c. CASE 3 – FOUR-LOOP REACTOR VESSEL

NOTE: ALL DIMENSIONS WERE OBTAINED FROM GENERAL ARRANGEMENT DRAWINGS AND ARE CONSIDERED REFERENCE. ACTUAL DIMENSIONS WILL BE ON A PLANT-BY-PLANT BASIS.

Figure 5. Thermal Annealing Zone-Cases 1 to 3.

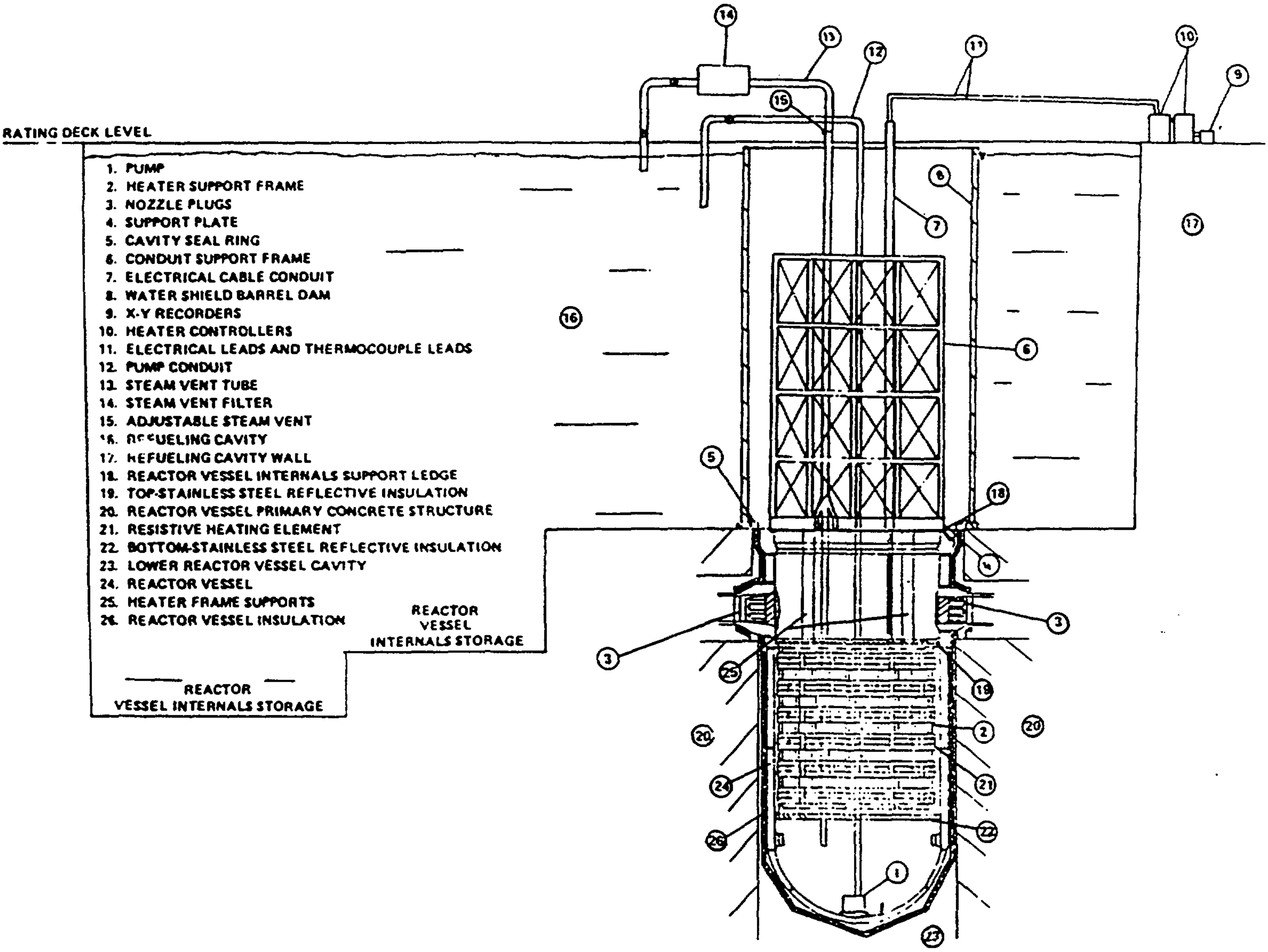


Figure 6. Design of Conceptual Thermal Annealing Apparatus.

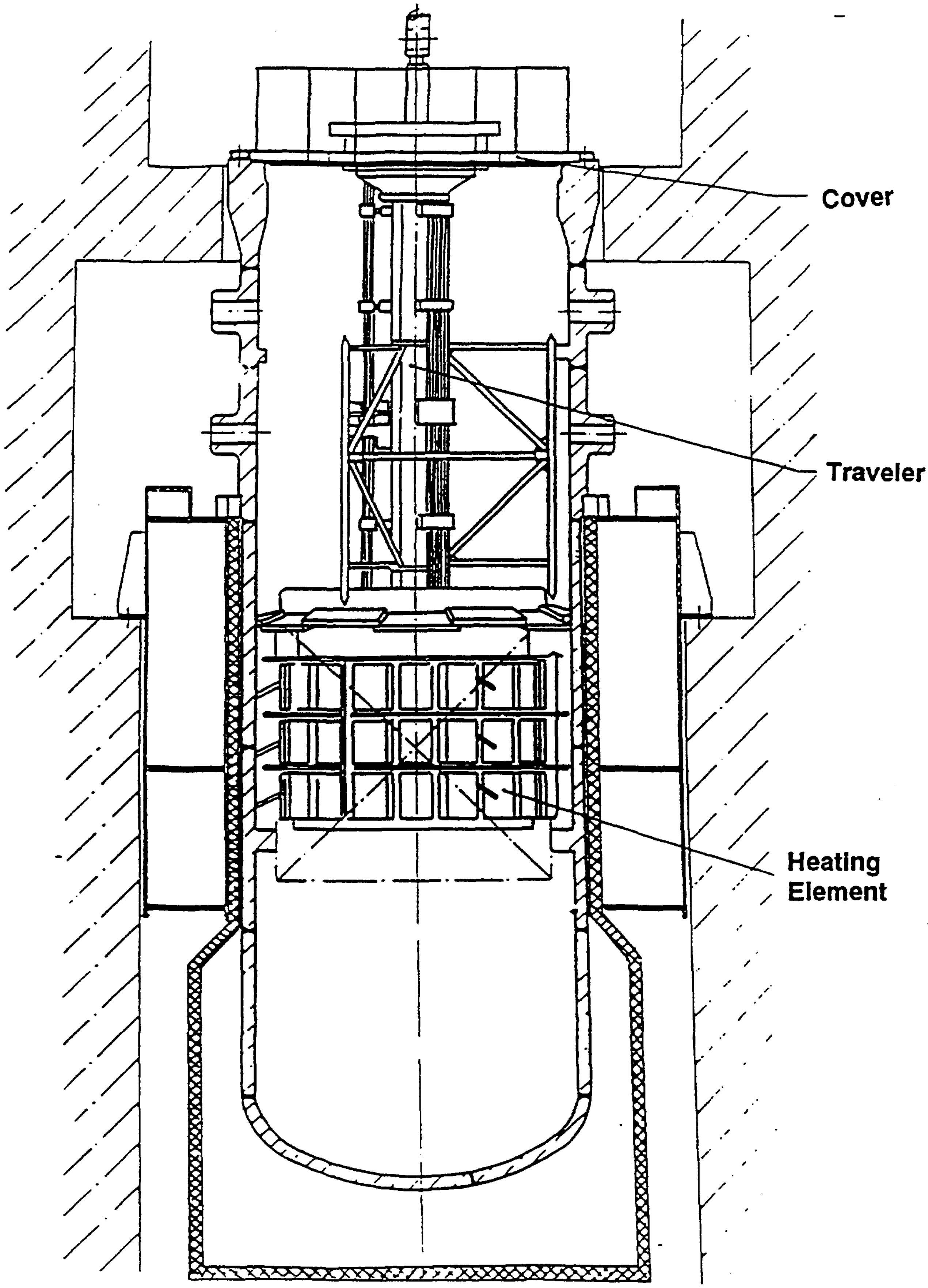


Figure 7. Cross Section of the Heating Element Installation

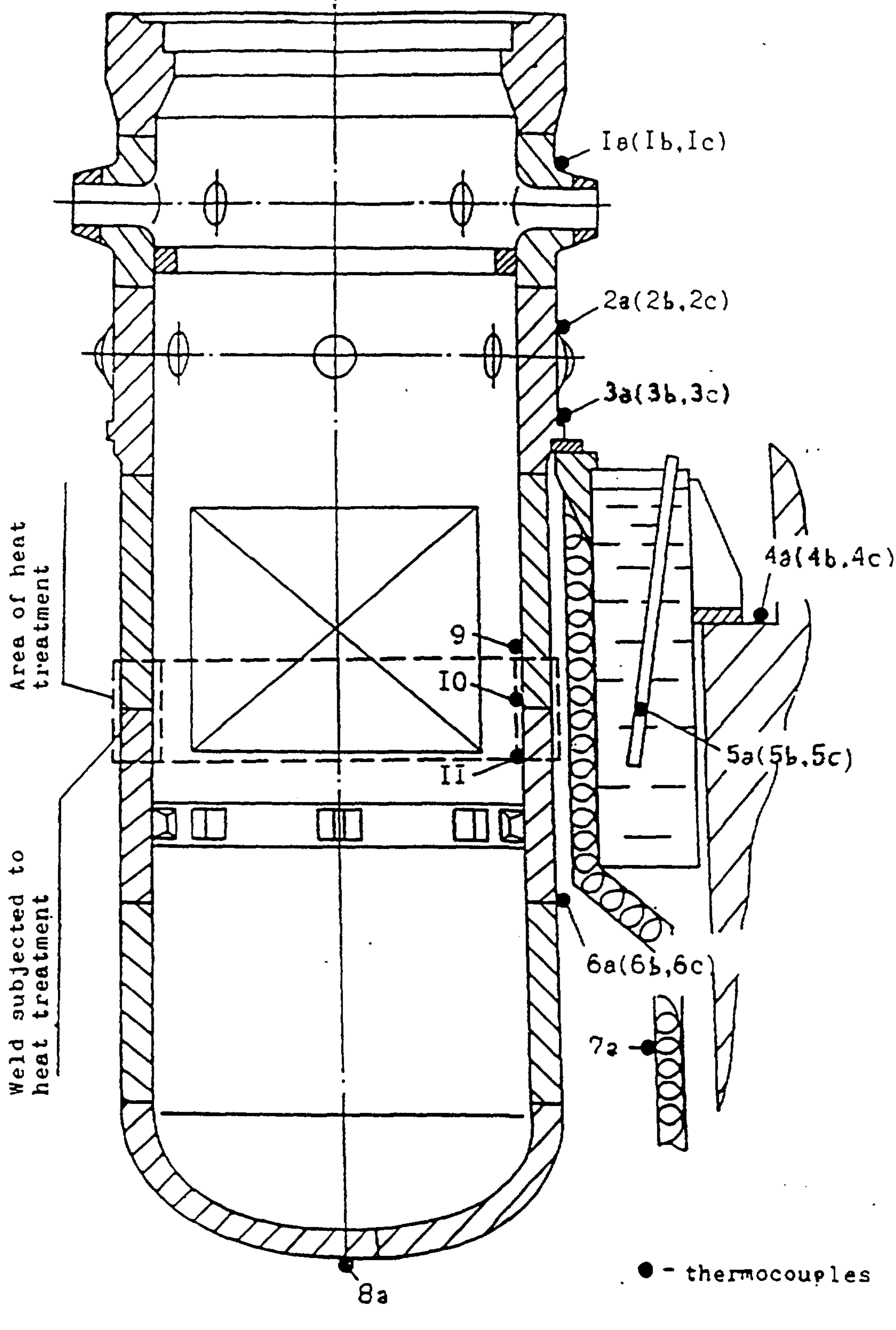


Figure 8. Arrangement of Thermocouples on the Vessel and Reactor Pit Equipment.

Pos. I-8 of the Figure Correspond to Additionally Installed Thermocouples; No. 9-II – to Thermocouples of the Heating Apparatus.