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Summary

This report summarizes the results of the HTR-M investigations on vessel materials.

The technological review and data base work identified the available information and data to be incorporated and confirmed the choice of Mod 9Cr 1 Mo steel as the material to use in the irradiation test programme. Properties for inclusion in the database are assembled to provide a platform of materials information for technological development. A recommendation has been made that thermal ageing should be investigated in the future.

The fabrication difficulties experienced in manufacturing the thick section welded features confirmed the need for the R & D work carried out within HTR-M. Optimisation work was needed to overcome the cracking found in the welds and to validate Mod 9Cr 1 Mo steel as a vessel material. The results also confirms that the levels of irradiation to be experienced by the HTR reactor Pressure Vessel was small compared with batch-to-batch variations. The assessment showed that the RCC-MR weld factor Jr = 0.9 for Mod 9Cr 1 Mo steel weldments was applicable to these RPV weld tests. Recommendations have been made with regard to further actions in future programmes to advance the material qualification and to widen the materials database to cover additional temperatures and conditions.

1. Introduction

The need for reliable material property data is a key issue in the development of the HTR and VHTR technology. The development of advanced HTR concepts requires an understanding of material behaviour plus materials data obtained under representative reactor operating conditions for the main HTR components important to safety and feasibility. The HTR-M project has as one of its objectives the improvement of the reactor pressure vessel (and other pressure vessels used to contain primary circuit components) for future HTR's. This work is covered in Work Package 1 of the HTR-M project and it is anticipated that improvements will be found on the basis of progress in the related areas of design analysis, structural integrity analysis, provision of materials property data and testing of pressure vessel welds under representative conditions.

This document contributes to the 5th Framework Project of HTR-M and provides a summary of the results obtain on Reactor Pressure Vessel materials (Deliverable 1.6).

2. HTR-M Vessel Work Objectives

Work Package 1: Reactor Pressure Vessel (RPV):

The work is contained in four individual tasks and involves the following two activities:

- Review existing materials used in gas-cooled reactors and previous high temperature reactors plus materials database on design properties (which will lead to identification of data omissions for future R& D testing)
- Carry out specific tests on welded joints under irradiated and non-irradiated conditions.

Fabrication and the effects of irradiation/environment on weld behaviour are considered crucial issues in determining RPV structural integrity for future HTR application.

The project resources allowed one type of vessel steel to be irradiated. A recommendation was made, on the basis of the review covering operating conditions, available materials data and fabrication details to test Mod 9Cr 1 Mo steel welds under irradiated and non-irrdaiated conditions.

The list of RPV deliverables provided by the HTR-M project are given in Table 1. A list of reports produced for the Project on RPV material activities is given in Table 2.

3. Results of Tasks 1 to 4

3.1 Task 1 - Review of RPV materials and design

3.1.1 Materials review

The technological review (Ref. 1) considered the materials used for developing reactors as well as experience from other reactor systems (see Table 3). The RPV forms part of the pressure boundary that contains the helium coolant during normal and abnormal conditions and provides structural support and alignment for the components within it (core and core support structures). Assurance of safety is of key importance and HTR vessel materials can operate at temperatures higher than 450^{o}C. Information is needed on materials behaviour during manufacture, in the operating environment (neutron-irradiation, operational temperature, and helium environment) and on welding for assurance of integrity.

Two types of materials are currently used for RPV's, depending on the temperature of the pressure boundary under normal operation or design conditions:

- C-Mn or C-Ni-Mo steels such SA 508 and its European derivatives. These are used for the 'cold' (cooled) vessel concept similar to that adopted for the PBMR reactor. The advantage of these steels is the large experience that underpins LWR vessel technology and design rules.

- Cr-Mo steels with modified 9Cr1Mo steel grades similar to ASME Grade 91 being the prime candidate: These are used for the 'warm' vessel concept such as adopted for some designs, particularly the GT-MHR reactor which is intended to operate at 400 – 460°C with potential temperature excursions up to 570°C.

These two types of steels have similar strength levels at temperatures up to 370°C but above 450°C allowable stresses for all materials fall off rapidly. The use of modified 9Cr 1Mo steel allows higher temperatures with only a gradual reduction in design strength at temperatures up to 450°C (Fig. 1).

For the HTR-10 test reactor in China, silicon killed C-Mn steel to ASME SA 516-70 is specified which is similar to the C-Mn steels used in the UK for the construction of both steel and concrete pressure vessels in CO_2 cooled reactors of the Magnox and AGR types. For the HTTR the RPV temperature is close to the core inlet temperature of 400°C and therefore 2¼Cr 1Mo steel has been used for the vessel.

Modified 9Cr 1Mo steel has been specified as having the best potential for both large and small reactors with the belt-line region of the vessel running at the core inlet temperature. This material is not yet qualified for pressure vessel use according to most design codes. The RCC-MR design rules for steam generator tube material in modified 9Cr 1Mo steel were extended during the European Fast Reactor (EFR) project to cover its use for tubeplates, shells and hemispherical ends in steam generators with outlet steam temperatures near 500°C. Comparison of estimated primary stress intensity values for GT-MHR with EFR values of S_{mt} show a close consistency. It should be noted however that Russian design rules use different factors in deriving design stresses from specified mechanical properties and therefore may not indicate the same level of similarity.

Another less attractive option is the 12Cr steel adopted in Japan for Fast Reactor steam generator applications and used extensively in steam plants in Europe. For this material welding is more difficult (i.e. careful control of temperature is required during preheat, inter-pass and post weld heat treatment) and although some thick section components have been made, most of the service experience relates to thinner sections (pipes, headers). Although 12Cr has a slightly higher strength (10%) below 500°C (where the design is governed by short term tensile properties) above approximately 500°C creep behaviour is more important, and here modified 9Cr 1Mo steel has superior creep strength.

3.1.2 Compatibility

For material compatibility, the main concerns are on metal loss/ carburisation/ decarburisation due to the effects of the impurities in the helium coolant. HTR gas chemistry is dominated by the impurities H_2, H_2O, CO and CO_2. Low concentrations of H_2O and H_2 are produced by leakage and / or de-sorption from both metal surfaces and graphite. CO, CO_2, and CH_4 may be produced by coolant / graphite reactions at high temperatures.

Information on He impurity levels has been gathered from various sources and thermodynamic assessments made for these atmospheres. Values of the carbon activity and oxygen potential (notional partial pressure of oxygen) were plotted on metal-gas equilibrium diagrams for iron and chromium. Points were included for identified HTR cases and typical AGR gas compositions. Temperatures relevant to the PBMR and GT-MHR vessels were considered.

For the HTR vessel, metal losses will be controlled by either carburisation in cold wall vessels, or decarburisation combined with internal oxidation in warm wall vessels. The kinetics of these processes will determine the effective metal loss. Overall except for the possibility of carburisation at cold wall vessel temperatures, coolant compatibility issues are not expected to have a strong direct impact on materials selection because metal loss allowances will be a small proportion of the pressure vessel wall thickness.

3.1.3 Irradiation & ageing

The main safety and structural integrity concerns for the RPV are at the vessel welds, at thicker sections, hot spots, the belt line and regions important to functionality. Fracture, fatigue and creep-fatigue (depending on temperature) are the main damage mechanisms and degradation mechanisms (irradiation, thermal ageing, temper-embrittlement and corrosion) the main environmental considerations. The effects of irradiation on the toughness of LWR steels have been studied extensively, the major effect being an increase in the ductile brittle transition temperature (DBTT) due to the combined effects of matrix hardening and grain boundary weakening. This is often associated with the residual elements phosphorus (P) and copper (Cu). For LWR this threat is generally considered to be significant when irradiation causes a change from ductile to brittle fracture at temperatures within the operational envelope. For the warm vessel option, no data are available for modified 9Cr1Mo steel irradiated under conditions expected for the HTR. Information has however been obtained at much higher doses (=1dpa) (Ref. 7) and these show a strong effect of irradiation temperature in the range 250-450°C. Effects may be small on ?DBTT at 400-440°C but measurable at 350°C (Fig. 2). There is no information on the behaviour of modified 9Cr 1Mo steel welds.

On thermal ageing embrittlement (TAE) the base materials and weld metals of both cold and warm HTR vessel steels normally have very fine grain sizes and therefore the most likely embrittlement location is the coarse-grained heat affected zone. This is a narrow region adjacent to the fusion boundary where temperatures of 1100 to 1300°C are reached during welding. Cold vessels operate at temperatures below the embrittlement range (350-550°C) so thermal ageing embrittlement is only possible during transients involving exposure to higher temperatures for tens or hundreds of hours. “Warm” vessels on the other hand operate in the TAE temperature range throughout their lifetime and therefore TAE is of greater concern with regard to vessel integrity. For a material such as modified 9Cr 1Mo steel operating in a “warm” vessel environment, thermal ageing effects in 40 years at 450°C may be significant but useful experiments on this are not be possible within the time scale of HTR-M.

3.1.4 Material selection

The above considerations and the state of the art review (Ref. 1 & 2) confirm modified 9Cr 1Mo steel as having the most uncertainties as to the available database and fabrication experience with respect to HTR vessel characteristics. This material also potentially has the wider applicability in terms of temperature operation. Since for the HTR-M Project it is only possible to perform irradiation tests on one material this represents the best candidate for the experiments. Modified 9Cr 1Mo steel was therefore selected for the specific experiments on welded joints under irradiated and non-irradiated conditions proposed in Task 3.

3.2 Task 2 - Materials properties database

An important objective of the HTR-M project was the development of a database of materials information for the key components (Ref. 3). For the Reactor Pressure Vessel, prime candidates for inclusion were C-Mn steels (as used in the UK Magnox and AGR types), SA 508 Grade 3 Class 1 (LWR) or its European equivalent, 2¼ Cr-1Mo steel as used on HTTR and modified 9Cr1Mo. 12Cr steel was also included although because of poor welding capabilities this material is unlikely to be a serious contender. Aspects such as composition, manufacturing information, test and design data plus environment were included in the design of spreadsheets. The materials property database includes code data where available, and raw data for analysis. Significant amounts of a data exist for all these steels and the main focus is on comparisons and assembling relevant design data.

In addition to vessel steels, the database covers high temperature alloys and graphites. The overall layout proposed for the database covers material designation, provisions, products & parts, test data and design property information. The design property information is an important output from the database as its expected to help drive future decisions on information needs and tests. Design needs plus omissions and shortage of test data are expected to be the basis on which the database will be built up and expanded.

The Alloys-DB system developed at JRC as a web based system for high alloy was investigated as a potential means for housing the information on the web in the long term to allow remote partner access and to maintain secure transfer of information between the different partners and countries. A PC version of Alloys-DB was provided in order to assess its suitability and agreements were made to allow access by all the HTR-M participants. Following a series of presentations and discussions on the potential benefits/ drawbacks of Alloys-DB at progress meetings it was agreed that the transfer and assembling of the available database of information would be done in two stages:

1. Collection of the information on to a CD-Rom via the co-ordinator.
2. Provision of the information to JRC for transfer on to the Alloys-DB system.

HTR relevant data previously established on the Alloys-DB system by JRC and FZJ would also be available to the project.

A document dealing with the management and transfer of the information provided by the HTR-M partners has been issued, also a series of datasheets compiled and loaded onto the SINTER website and on to the CD-Rom of the project. The relevant supporting technical documents relating to this task are listed in Tables 1 and 2. Subsequent information from the HTR-M1 project and future European VHTR activities will be added to the core of data provided by the HTR-M project.

3.3 Task 3 – Tests on welds under irradiated and non-irradiated conditions

This task deals with the experimental investigations performed on Mod. 9Cr 1Mo steel welds to establish test information and the behaviour of the material under HTR typical radiation conditions.

(i) Procurement & fabrication of welded test pieces

Task 3 required the procurement of a sufficient amount of material (rolled plate) for the manufacture of a number of representative vessel weld features. This was done by Framatome who also selected and purchased a reduced phosphorus filler material for the fabrications from the producer Kobe steel. Framatome had tested several filler materials and this type of filler had been successfully used in the past for the producing 30mm thick welded joints with a large range of welding parameters. It was considered the best available for the application. The chemical analysis of the filler material was as follows:

C	Mn	P	S	Si	Ni	Cr	Mo	Nb	V	N2	Cu
0.08	1.00	0.004	0.005	0.18	0.71	9.03	0.90	0.04	0.18	0.0189	0.17

For the tests Framatome manufactured representative thick section welds using the narrow gap TIG weld process with a double V weld preparation. The post weld treatment used was 10.5h at 760^{o}C. The total weldment length was approximately 1m. The first pilot test welds (150 mm thick) showed a significant number of micro-cracks (24 over a area of 6600mm^{2}) in the weld metal. These cracks (< 0.3 mm) were not detected during X-ray or ultrasonic examination but seen when performing a micrographic examination. The shape of the cracks indicated hot cracking. This was unexpected as the selected filler had the lowest tendency to hot cracking among the products tested by Framatome. The fabrications were repeated using a lower energy weld sequence and although there was a significant reduction the problem was not completely solved. The number of micro-cracks was reduced to 5 (see Fig. 3). Changing the welding parameters further was not considered viable if the tests were to mirror industrial practice as far as possible. Enquiries and discussions established that

- Procurement of a new filler material would take the best part of a year
- There would be no assurance of success or that a new filler would be available commercially.

It was therefore agreed to adopt the Kobe filler with reduced energy for the tests. It was expected that the HAZ would be defect free and the density of the cracks in the weld metal would allow machining of defect free specimens to be achieved. A procedure was developed by Framatome in co-operation with JRC and NRG for identifying cracks, machining specimens and checking them.

This work represented the first trials on this material for such thick section vessel type welds and considered only one geometry (Ref. 8). The trials were therefore a first step in confirming the fabrication parameters for such an application. Later developments in the fabrication procedure will need to include forged parts. Subsequent work was continued within Framatome to explain the reasons for the phenomenon and also to help with the selection of a future filler material that would avoid this problem (Ref. 9). These trials identified sulphur levels to be important also the manganese to sulphur ratio and tests on a

200mm joint confirmed this. Further trials on submerged Arc Welds (SAW) and Shielded Metal Arc Welds (SMAW) identified suitable filler materials although for the SMAW (which would be used for repair) further investigations are underway to improve impact properties.

The presence of the hot cracks within the specimens necessitated the development of an (ultrasonic) inspection procedure to ensure only sound material was used for the irradiation tests. Unfortunately after inspection and machining of the specimens it was not possible to fill all the material locations available in the Lyra Rig. The small number of spaces not occupied by weld materials was filled with P91 grade parent plate steel to provide a notional comparison of parent plate behaviour.

3.3.1 Test Programme & Reference Testing (Ref. 5)

A test Programme was agreed consisting of tensile and creep tests (see Table 4) with tests jointly performed by NRG (irradiated) and JRC (non-irradiated). The objective of the programme was to measure the effects on properties of a typical HTR vessel End of Life (EOL) neutron dose with a relevant neutron spectrum. The EOL neutron dose depends on the design and type of core. The nominal dose of 8 x 10^{22} n/m^2 (E>0.1 MeV) was agreed for planning purposes. The nominal irradiation temperature and main test temperature was chosen as 375^{o}C, corresponding to the lowest expected operating temperature for a modified 9Cr vessel.

a) Preliminary Tests

As two organisations (NRC & JRC) were involved in the testing, comparison tests were carried out on modified 9Cr1 Mo steel to check the testing practices of the two partners, especially for tensile and fracture toughness testing. The results confirmed the similarity in test procedures and gave confidence in the comparability of the data.

b) Weld & Parent plate reference tests

Tensile, creep, impact and fracture toughness tests were carried out at JRC. Tensile tests were performed at Room Temperature (RT) and temperatures up to 550^{o}C. Tensile and creep data were obtained on base material specimens (longitudinal and transverse(L and T) directions), all weld metal specimens (WM) and cross weld (CW) specimens.

Impact and fracture toughness tests were carried out with notches in the weld metal (WM), heat affected zone (HAZ), and base metal in two orientations (L-T and T-L). Impact test temperatures as low as –150°C were used to determine ductile to brittle transition temperatures (DBTT) while fracture toughness tests were performed at room temperature and 375°C.
The creep tests were carried out at 550^{o}C with target rupture times of 100, 300, 1000 and 3000h. The sequencing/ planning based on a proposal by NNC worked well except for the cross weld tests. Comparisons were made with CEA data and also with non-factored RCC-MR values.

Impact values were determined using sub-size and full size specimens, the latter to study size effects. Again the same nominal four material conditions were used (WM, HAZ, L-T, T-L) for the sub sized tests, three for the full sized tests. Impact test temperatures as low as –150°C were used to determine ductile to brittle transition temperatures (DBTT).
For the fracture toughness tests, results were obtained at RT and 375^{o}C (irradiation temperature) again in the four conditions, with the notch placed in the HAZ or WM as

appropriate. For evaluation the actual measured tensile data were used in calculating the full J-R curves.

3.3.2 Irradiation and Post Irradiation Testing (PIT) (Ref. 6)

The objective of the irradiation experiment was to achieve the End of Life (EOL) neutron dose with a relevant neutron spectrum for the HTR pressure vessel at the nominal operating temperature. The EOL neutron dose depends on the design and type of core. The conditions achieved were as follows:

- Average neutron fluence – 9.44×10^{22} n/m^2 (E>0.1 MeV), 5.85 mdpa
- Maximum neutron fluence 11.41×10^{22} n/m^2 (E>0.1 MeV), 7.07 mdpa
- Minimum neutron fluence 7.97×10^{22} n/m^2 (E>0.1 MeV), 4.93 mdpa
- Nominal irradiation temperature - 375^oC +/- 10^oC
- Irradiation to take place in the poolside of HFR.

Equipment design for the irradiation experiment was based on previous pool side irradiations performed at NRG using the LYRA Rig, which allows accurate control of temperatures in inert atmospheres. Figure 4 shows details of a typical loading. Tungsten plates shield the specimen holder from gamma radiations to avoid high gamma heating. Heaters in the rig are used to maintain nominal temperature conditions. The difference in temperature between specimens is maintained within 15^oC.

The actual temperature of the specimen was monitored using thermocouples at different levels within the irradiation rig. Neutron fluences were monitored using pieces of nickel-cobalt, titanium, iron and niobium to be analysed after the irradiation. Fluence gradients due to the flux buckling and self-shielding were determined using the neutron monitors. The specimen fluence gradients were kept as low as possible (<5%) with the notch axis parallel to the HFR core box. Pure helium was used to provide the inert environment in the test.

The irradiation exposure was completed over a two-month period during 2 reactor cycles. The capsule was rotated through 180° after the first reactor cycle to reduce fluence variations,

The temperature of irradiation was 375^oC with a deviation of up to 11^oC (flux buckling). The cooling period after irradiation was longer than planned due to HFR maintenance activities.

PIT included tensile, fracture toughness, impact and creep tests all within NRG shielded facilities. The tensile tests were performed using an electro-magnetic test machine with a 10KN load cell and split three zone furnace located in the shielded hot cell. No extensometry was used, elongations being determined on the basis of cross head movement and the original gauge length. The plastic strain measured after fracture conformed with the recorded strains. 0.2% yield stress, UTS, fracture elongation values were determined from the recorded stress-strain curves. After testing some of the small identification numbers were unreadable due to irradiation and heating during the testing and only the results where the numbers were still readable could be assigned.

For the creep tests three shielded creep machines were used covering all four conditions (approx 100, 300, 1000, 3000h = ~ 4500h) with testing taking around 250 days. A split three-zone furnace with individual temperature control of each section was used to heat the specimens to 550^oC. For the impact tests a small 50J pendulum machine was installed in a hot cell. The testing was completed within 3s of leaving the temperature-conditioning chamber, and compensation made for temperature loss using a spot weld thermocouple. As with the reference tests, for each specimen the instrumented data was used to determine the DBTT and USE values. The fracture toughness tests were carried out at RT and 375°C and the results used to construct J-Δa curves.

3.3.3 Results

a) The tensile test results are shown in Table 5. In general the values shown in Table 5 are the average from two tests where strength values differed by less than 15 MPa and ductility values were within 2%. One weld metal specimen extracted from a different part of the joint (Plate 4, closer to the weld root) gave a significantly different yield strength to its duplicate and other specimens from Plates 5 and 6. The result from Plate 4 has been omitted from Table 5 but may indicate that properties vary with location within the joint.

With the above reservation the results are internally consistent and much as expected on the basis of other data from similar steels. The following points can be noted:

- There was a slight increase in room temperature yield strength as a result of the irradiation but there was no increase in the 550°C yield strength (annealing at the test temperature).
- Of the welded specimens the weld metal had the highest ductility, the cross weld specimens the lowest.
- As temperatures increased the yield to ultimate strength ratio increased (especially between 450 and 550°C).

b) The rupture data are shown in Figure 5. The creep rupture strengths for the two base material orientations were almost identical. There was no significant effect of irradiation on the rupture strengths of either the base material or weld metal. Figure 5 therefore shows all the base metal and all the weld metal data without distinguishing orientations or irradiation.

Rupture times for the base material are close to the RCC-MR minimum values. The weld rupture lives at high stresses are significantly lower than the base material and RCC-MR minimum lives (by a factor of 3-5 at around 230Mpa). At the lowest stresses investigated in this work (170-175 MPa) both base material and weld metal lives reached or exceeded the RCC-MR minima. The creep strain data obtained during the tests showed trends consistent with the rupture data while the rupture ductility (%El and %R of A) was high and unaffected by irradiation.

The only systematic influence of irradiation was seen in the cross-weld tests where testing times were longer and minimum strain rates lower. The failure location for all the non-irradiated tests creep tests for the CW specimens was in the weld metal. In the irradiated specimens there was a change in failure location from weld metal at higher stresses (213 and 190 MPa) to the intercritical/overtempered HAZ at lower stresses (186 and 179 MPa). This change in failure location is consistent with the improvement in the weld metal rupture strength relative to the base metal strength at longer times. It is not clear why this change did not occur also in the tests on the unirradiated welds.

It can be concluded that that, overall the irradiation was not shown to have an adverse effect on creep properties but that the rupture strength of welded joint and its constituents was at or below the RCC-MR minimum. Further tests are needed to check whether the trends noted above continue in the longer term i.e.

- An increase in the strength of base material, weld metal and cross weld strength relative to the RCC-MR minima.
- Change to failure in the intercritical/ overtempered HAZ in both unirradiated and irradiated cross weld specimens.

c) The impact tests on unirradiated sub-sized specimens showed that the lowest ductile brittle transition temperatures (DBTT) occurred in the HAZ specimens and in the weld material. Both plate directions showed similar DBTT results. The weld metal also showed the highest upper shelf energy (USE) confirming the high upper shelf fracture toughness of this material.

The results for irradiated sub-sized impact specimens again showed the HAZ of the weld to give the lowest DBTT as with the non-irradiated tests. The irradiation shift in DBTT (?DBTT) ranged between 5 and 12°C when based on impact energy. This range and the associated dpa range are plotted on Figure 2. It appears that the materials in the test weld are less susceptible to embrittlement than other Cr-Mo steels (including HT 9) at least for irradiation to around 10 mdpa. In addition an increase of 20-27% in the upper shelf energy was noted. This was accompanied by a loss of upper shelf lateral expansion of 8 to 20%.

The specimen size effect was studied in unirradiated specimens by obtaining full sized charpy curves for the base metal and weld metal only. An evaluation against the sub-sized results showed the scaling factor to be consistent with previous reported values for 9Cr steel.

d) In the fracture toughness tests on irradiated material, the highest measured toughness values were obtained on the weld metal. Values ranged from $J_{0.2bl}$ = 251 kJ/m^2 to 468 kJ/m^2 at 375°C and $J_{0.2bl}$ = 488 kJ/m^2 to 833 kJ/m^2 at RT. The two base metal orientations gave somewhat lower toughness values. A reduction of fracture toughness with a factor 40-50% was observed between tests at room temperature and tests at 375°C. No conclusions on irradiation effects on fracture toughness can be drawn, since no reference results are available at this moment. The toughness values are size dependent and only valid for specimen thickness of B=10 mm.

In general it can be concluded that for the end-of-life dose anticipated for HTR, the irradiation effects on mechanical properties are minimal. The main concerns arising from this work are not related to irradiation effects i.e.

- Hot cracking of the weld metal.
- The relatively low creep strength of the base metal and weld metal.

3.4 Task 4 - Synthesis & Evaluation of weld factor (Ref. 4)

This task was intended to cover the synthesis (this document) and determination of the weld stress rupture factor from the above results in accordance with the requirements of the RCC-MR code.

An assessment of the weld factor (Ref. 4) was made by comparing the weld metal and cross weld results with the base material results. The assessment showed that except for one test of 25h lifetime, the results were consistent with the = 0.9 for Mod 9Cr 1 Mo steel weldments. The RCC-MR value of Jr was previously derived from tests on much thinner section welds within the European Fast Rector (EFR) programme.

A comparison was also made between the stress rupture data from this study and non-irradiated results from a plate supplied to CEA. This used a slightly lower PWHT temperature of 750°C compared with the 760°C value adopted for the HTR-M plates. The assessment showed that the low dose encountered by the HTR vessel has no important influence on the mechanical behaviour. The lower PWHT temperature used for the CEA plate however seemed to result in a higher 550°C creep strength, and this requires further investigation.

4. Summary of Main Findings

The objective of the HR-M project was to improve knowledge of materials for future HTR's. For the reactor pressure vessel, the main findings and areas where further investigations are needed are as follows:

4.1 Literature Review & database:

- The vessel review work has focused on materials used in new and developing plants and shown that two types of steel are currently favoured for the Reactor Pressure Vessel (RPV):
    - Cooled vessel - Mn-Ni-Mo or C-Mn steels, SA508 or equivalent (LWR steels)
    - Non cooled (warm) vessel - Cr-Mo steels – $2^{1}/_{4}$ Cr 1Mo; Mod 9Cr 1 Mo steel is an emerging prime candidate
- Mod 9Cr 1 Mo steel is not yet qualified as a pressure vessel steel but offers a potential for the RPV to operate at higher temperatures with only a gradual reduction in strength up to 450°C.
- The state of the art review confirmed the selection of Mod 9Cr 1 Mo steel for the Irradiation test programme
- Considerations need to be given to the cooled vessel option to cover transients involving temperatures exceeding the PWR design limit of 370°C.
- Actions need to be put in place to investigate HTR vessel manufacturing technology and design rules with respect to:
    - Ring forging capabilities for the cooled and non-cooled HTR vessel diameters and geometries.
    - Long term creep tests at 400-500°C to resolve uncertainties on design stresses and weld factors for warm vessel materials.
    - Engineering assessments / experiments to establish progressive deformation/ ratcheting rules for warm vessels in cyclic softening modified 9Cr-1Mo steel.
- No data were found on irradiated thick section modified 9 Cr 1 Mo material welds relevant to the HTR vessel application.
- It was recommended that Mod 9Cr 1 Mo tests are extended to cover the effects of thermal ageing on the coarse grained region of the weld heat affected zone.
- HTR gas /metal reactions are likely to lead to carburisation in cold wall vessels and decarburisation combined with internal oxidation in warm wall vessels. Because of the thick sections these will only become significant if unstable processes occur (such as metal dusting). The sensitivity to helium composition needs to be checked.

4.2 Test piece fabrication & testing:

- The Kobe Filler selected as best available from FBR and Japanese experience was successful on thinner (40 mm) sections but resulted in hot micro-cracking 1n 150 mm thick narrow gap welds.
- The final joints were fabricated using narrow gap TIG, Kobe steel filler metal and low welding energy (to minimise cracking). Deviations between this and final production practice were not expected to yield significantly differing properties and derived factors for effects of welds and irradiation are unlikely to be affected.
- Investigation of sensitivity of creep strength to PWHT temperature requires further investigation
- Very limited information was available on vessel transients and environmental conditions on which to base the test conditions.
- Further investigations of hardening / softening characteristics of the welded joint and on complete weld behaviour are needed. Creep/fatigue interaction effects should be included.

4.3 Effects of irradiation

- HTR-M PIT has provided results at one creep temperature (550^{o}C). Further tests at lower temperatures are needed in the 6th Framework Programme (6FP) to widen the database.
- PIT tests has shown that the low end of life dose encountered by the HTR vessel has no important influence on the mechanical behaviour of the weld at 375°C except for the (unexplained) acceleration of failure location transition in cross weld tests from the weld metal to the HAZ
- The comparatively short term tests gave results in line with the RCC-MR weld factor Jr = 0.9 for Mod 9Cr 1 Mo steel weldments. No significant influence of irradiation on Jr was encountered.

5. Conclusions

The vessel work provides a review of available information and properties on steels used in both existing and new and developing plants. SA 508 type steel is proposed for the cooled design of the current PBMR and modified 9Cr 1Mo steel for the non-cooled (inlet temperature) vessel design of the GT-MHR. Available properties on both types of material were included in the database. Because Modified 9Cr 1Mo steel has the wider applicability, the greater uncertainties on database and the least fabrication experience, the irradiation experiments were performed on Modified 9Cr 1Mo steel welded joints.

Following material procurement (rolled plate) and (production shop) welding trials a number of representative thick section vessel weld features were produced and the required test samples (weld metal, heat affected zone, base material) machined. The welding trials were a first step in confirming the fabrication parameters. They showed the importance of optimising the filler material composition and welding procedures to avoid cracking and optimise strength.

After irradiation in the pool side of the High Flux Reactor (HFR) for two complete loading cycles (average neutron fluence – 9.44x10^{22} n/m^{2} (E>0.1 MeV), equivalent to 5.8 mdpa) at a temperature of 375^{o}C. Post Irradiation Testing (P.I.T.) was carried out to determine tensile, creep, impact and fracture properties. Comparisons with equivalent non-irradiated test results showed that for all the material conditions investigated, the irradiation effect was small compared with batch-to-batch variations. The assessment also confirmed that the RPV weld stress rupture results were consistent with the RCC-MR weld factor Jr = 0.9 for Mod 9Cr 1 Mo steel weldments.

Recommendations have been made with regard to further actions in future programmes to advance the material qualification and to widen the materials database to cover additional temperatures and conditions.

6. References

1 R.D.W.Bestwick
State of the art on HTR vessel materials
Report No. HTR-M 02/12 D 1 1 49

2 M.T. Cabrillat & A. Buenaventura
Recommendations for RPV design, choice of material, defect assessment and leak before break methodology
Report No. HTR-M 02/8 D 1 1 40

3 A.A. Braithwaite
Data base on pressure vessel steel plate and weld material for reactor pressure vessel (RPV)
Report No. HTR-M 02/11 D 0 0 50

4 Dubiez-le Goff
HTR-M Vessel materials – evaluation of weld factor
Report No. HTR-M 04/10 D 1 4 98

5 J. Hegeman & H. Rantala
Results of testing and irradiation behaviour of Mod 9Cr steel RPV welds and base material
Report No. HTR-M 04/10 D 1 4 99

6 J. Hegeman
Summary report on vessel materials
Report No. HTR-M 04/10 D 1 5 100

7 A.A. Braithwaite
Review of irradiation and thermal ageing effects in Mod 9Cr 1 Mo steel.
Report No. HTR-M 02/11 D 1 1 48

8 D. Pierron
Mod 9Cr 1 Mo steel – Manufacturing of Welded Joints
Report No. HTR-M 03/11 D 1 3 82

9 B. Riou, C. Escaravage, D. Hittner, D. Pierron
Issues in Reactor Pressure Vessel Materials, paper E16, 2nd Int. Topic Meeting, HTR2004, Beijing, China, 22-24 September 2004.
Report No. HTR-M 04/9 D 0 0 107

Table 1. HTR-M Deliverables list (WP1)

Deliverable No	Deliverable title	Month due	Current planned	Date Achieved	Main Organisation
1.1	Data base on pressure vessel steel plate and weld material for reactor pressure vessel (RPV)	24	01/11/2002	24	NNC
1.2	State of the art on HTR vessel materials	24	01/11/2002	25	NNC
1.3	Recommendations for RPV design, choice of material, defect assessment and leak before break methodology	24	01/11/2002	21	CEA/EA
1.4	Evaluation of factors for defect tolerance of RPV weld	42	01/05/2004	50	FRA
1.5	Complimentary irradiation data on base material and weld	42	01/05/2004	50	NRG
1.6	Summary report on vessel materials	48	01/11/2004	50	NNC

Table 2 HTR-M Project document list for WP1 - RPV material

HTR Report No.	Title	Author / Org
HTR-M-01/3-D-0-0-1	HTR-M Release Data	T.Austin / JRC
HTR-M-01/3-D-0-0-2	On-line Alloys-DB Access Agreement	T. Austin / JRC
HTR-M-01/9-D-1-1-9	HTR-M project - Preliminary review of RPV materials used on HTR and other gas reactors	R.D.W.Bestwick / NNC
HTR-M-01/7-D-1-0-15	HTR-M Work package 1 - Detailed description of tasks	H. Hegeman & B van der Schaaf/NRG
HTR-M -01/10-D-1-1-17	Material selection for HTR Pressure Vessel	A. Buenaventura/ EA
HTR-M 01/11 D.1.1.18	Report on Damage Mechanisms	M.T.Cabrillat/CEA
HTR-M 01/3 D.0.0.21	HTR-M Materials Data base - scope and proposed stages of development	D. Buckthope NNC
HTR-M 01/11 D.0.0.22	Scheme for HTR-M Materials data collection	D. Buckthorpe NNC
HTR-M 02/02 D.1.1.27	HTR RPV Materials Damage Mechanisms	A. Buenaventura EA
HTR-M 02/02 D.1.1.28	HTR RPV Sensitive Zones	A. Buenaventura EA
HTR-M 02/04 D 1.1.31	HTR - Identification and review of Pressure vessel candidate materials	C. Escaravage / Framatome-ANP
HTR-M 02/8 D 1 1 40	Recommendations for RPV design, choice of material, defect assessment and leak before break methodology	M.T.Cabrillat / CEA A. Buenaventura / EA
HTR-M 02/9 D 1 1 41	HTR Compared properties of Pressure Vessel candidate materials	C. Escaravage / Framatome ANP
HTR-M 02/11 D 1 1 48	Review of irradiation and thermal ageing - effects in modified 9Cr 1Mo steel	A. Braithwaite / NNC Ltd.
HTR-M 02/11 D 1 1 49	State of the art review of HTR materials	R.D.W.Bestwick/ NNC Ltd.
HTR-M 02/11 D 0 0 50	The HTR-M vessel materials database	A. Braithwaite / NNC Ltd
HTR-M 02/11 D 0 0 51	HTR-M vessel materials database Nov 02	A. Braithwaite / NNC Ltd
HTR-M 03/02 D 1 4 63	Specification of the test matrix for the RPV test programme	J. Hegeman / NRG
HTR-M 03/04 D 0 0 65	HTR RPV Materials. (Paper 3355). ICAPP 03 Conference, Cordoba, Spain, 4-7 May, 2003	A. Buenaventura/ EA

Table 2 (cont) HTR-M Project document list for WP1 - RPV material

HTR Report No.	Title	Author / Org
HTR-M 03/05 D 1 1 70	Effect of Environment on the material of GT-MHR Vessel	M. Cabrillat/ CEA
HTR-M 03/08 D 0 0 75	HTR-M metal materials database (BLANK)	A.A. Braithwaite / NNC Ltd
HTR-M 03/08 D 1 4 78	Proposed creep rupture test conditions for the HTR-M irradiation experiment on a modified 9Cr Mo steel weld	R. D. W. Bestwick/ NNC Ltd
HTR-M 03/11 D 1 3 82	Modified 9Cr 1 Mo steel, Manufacturing of Welded Joint materials	D. Pierron / Framatome.ANP
HTR-M 04/04 Q 0 0 85	Question sheet on Fluence levels and the energy spectrum expected for an HTR and VHTR in vicinity of reactor pressure vessel	D. Buckthorpe / NNC
HTR-M 04/05 D 1 4 88	Activation analysis – (up to 500 days after irradiation) of the Vessel steel Modified 9Cr 1 Mo for the LYRA irradiation.	J. Hegeman /NRG
HTR-M 04/10 D 1 5 97	Weldments for reactor pressure vessels – a guide to qualification, quality assurance and defect assessment.	B-C Freidrich/ Framatome.GmbH
HTR-M 04/10 1 4 98	HTR-M Vessel materials – evaluation of the weld factor	Dubiez-le Goff, Framatome.ANP
HTR-M 04/10 1 4 99	Results of testing and irradiation behaviour of Mod 9Cr 1 Mo steel RPV welds and base material.	J/ Hegeman/ NRG & H. Ranatla/ JRC
HTR-M 04/10 D 1 5 100	HTR-M Summary report on vessel materials	D. Buckthorpe/ NNC
HTR-M 04/11 M 0 0 101	Minutes of the 8th HTR-M plus M1 Progress Meeting at Framatome, Lyon	D.Buckthorpe/ NNC
HTR-M 04/10 D 2 1 102	Materials for Core & Internals: Report on results from HTR-M investigations on C/C composite material.	G. Dellette & R. Couturier/ CEA
HTR-M 04/10 D 2 3 103	Materials for Turbine: Report on results from HTR-M investigations.	R. Couturier/ CEA
HTR-M 04/10 P 0 0 104	48 Month HTR-M plus 36 month HTR-M1 Management Report	D. Buckthorpe/ NNC
HTR-M 04/12 P 0 0 105	Final Technical Report for the HTR-M Project	D. Buckthorpe/ NNC
HTR-M 05/1 D 2 1 106	Notes of a Technical Meeting with Dunlop on C/C Composites for Control Rod	A. Keenan/ NNC

Table 2 (cont) HTR-M Project document list for WP1 - RPV material

HTR Report No.	Title	Author / Org
HTR-M 04/9 D 1 0 0 107	Issues in Reactor Pressure Vessel Materials, paper E16, HTR2004 Conference, Beijing, China, 22-24 September 2004	B. Riou, C. Escaravage, D. Hittner, D. Pierron/ Framatome.ANP
HTR-M 04/9 D 1 0 0 108	Studies on Mechanical Behaviour of Mod 9Cr 1Mo steel, CEA r & d programme, paper E13, HTR2004 Conference, Beijing, China, 22-24 September 2004	M. T. Cabrillat et al/ CEA

Table 3. HTR Test Reactors and Designs for Future Plant

Country	Japan					South Africa	Russia + others	China	
System (1)	HTTR	MSF HTR	SSF HTR	HTR -GT 300	HTR -GT 600	PBMR	GT-MHR	HTR 10	MHGTR -IGT
Purpose (2)	Tests	H+E	E+DS	E	E	E	E	Tests	E+DS
Power, MWt	30	450-600	50	300	600	265	600	10	200
Core Type)	Block	Block	Block	P. Bed	Block	P. Bed	Block	P. Bed	P. Bed
He Pressure, MPa	4	6	7	6	6	7	8	3	6
Core inlet, °C	395	350 550	490	550	460	536	490	300 (8)	550
Core Outlet, °C	850 950	850 950	850	900	850	900	850	900 (8)	900
RPV wall, °C	400	-	-	<350	460	300	440	-	<350?
RPV ID, m	5.5	-	-	7.27		6.2	7.3	2.5	6
Beltline thk., mm (3)(4)	122	-	-	150	-	140	200 approx.	100 approx	-
Flange thk., mm (4)	300 appro	-	-	550 approx	-	365	670	-	-
Vessel mass, te (5)	-	-	-	800 (V)	<1000 (V)	600	1362 (V+C)	142	-
Material	2¼ Cr	Mod 9Cr	Mod 9Cr	SA508	Mod 9Cr	SA508	Mod 9Cr	SA516	-

Notes:

(1) Initials are as used in Reference 40

(2) E = electricity generation, H = hydrogen production, DS = de-salination

(3) BL = belt line, usually the minimum shell thickness. Max usually refers to that part of the cylinder containing the ‘cross vessel’ connection between the RPV and the power system vessel.

(4) Thickness is called ‘approx.’ when estimated from a small-scale drawing.

(5) V = vessel only - excluding bolted closure. V+C includes bolted closure, vessel mass about 1000 te.

Table 4 - HTR-M Test Matrix for Mod 9Cr 1Mo Steel Vessel Welds - Reference and Irradiation Testing

Specimen type	#	Conditions	Temperature.	Sub total	6th FP	Irr/ref	total
Small sized Charpy (KLST)	15	4		85		Irr & ref	170+
Tensile	2	4	4	32		Irr & ref	64
Creep	4	4	1	16	16*	Irr & ref	32 (32*)
Fracture Toughness (CT10)	5	4	2	20		Irr & ref	40**
Full size Charpy	11	3		25		Irr & ref	50***

+ 15 specimens in each plate direction (L-T, T-L), 15 specimens in weld metal, 40 specimens in the heat affected zone (HAZ)

*16 specimens will be irradiated in LYRA but tested in the 6TH Framework Programme (FP), also 16 additional specimens will be prepared for reference testing in the 6th FP

**4 additional CT specimens of commercial T91 for comparison

***25 additional full sized Charpy specimens will be included in the irradiation rig (6FP bending or impact tested in Eastern Europe through JRC-IE) and 25 specimens for reference testing, 12 for the base material (T-L), 11 with a notch in weld metal, 2 specimens with notch in HAZ (irradiation & reference)

Table 5 Tensile Test Results for Mod 9Cr Weld

Case	YS MPa	UTS MPa	%El	%R of A	YS MPa	UTS MPa	%El	%Rof A
	20 or 27°C				375°C			
BM/L	466	634	27	-	-	-	-	-
BM/L/I	506	648	27	76	414	516	20	75
BM/T	457	628	28	-	-	-	-	-
BM/T/I	498	644	28	75	408	513	22	74
WM	499	635	32	-	-	-	-	-
WM/I	515*	646	35	-	420	516	22	83
CW	454	624	23	-	-	-	-	-
CW/I	490	638	22	-	395	499	16	83
	450°C				550°C			
BM/L	-	-	-	-	331	353	30	-
BM/L/I	380	474	21	80	327	361	28	90
BM/T	-	-	-	-	-	-	-	-
BM/T/I	375	478	20	78	311	347	31	93
WM	-	-	-	-	-	-	-	-
WM/I	372	456	24	84	277	325	36	-
CW	-	-	-	-	-	-	-	-
CW/I	369	462	19	81	314	346	21	89

* A duplicate test from a different weld location gave a much lower yield strength

Figure 1. Allowable stresses for RPV materials

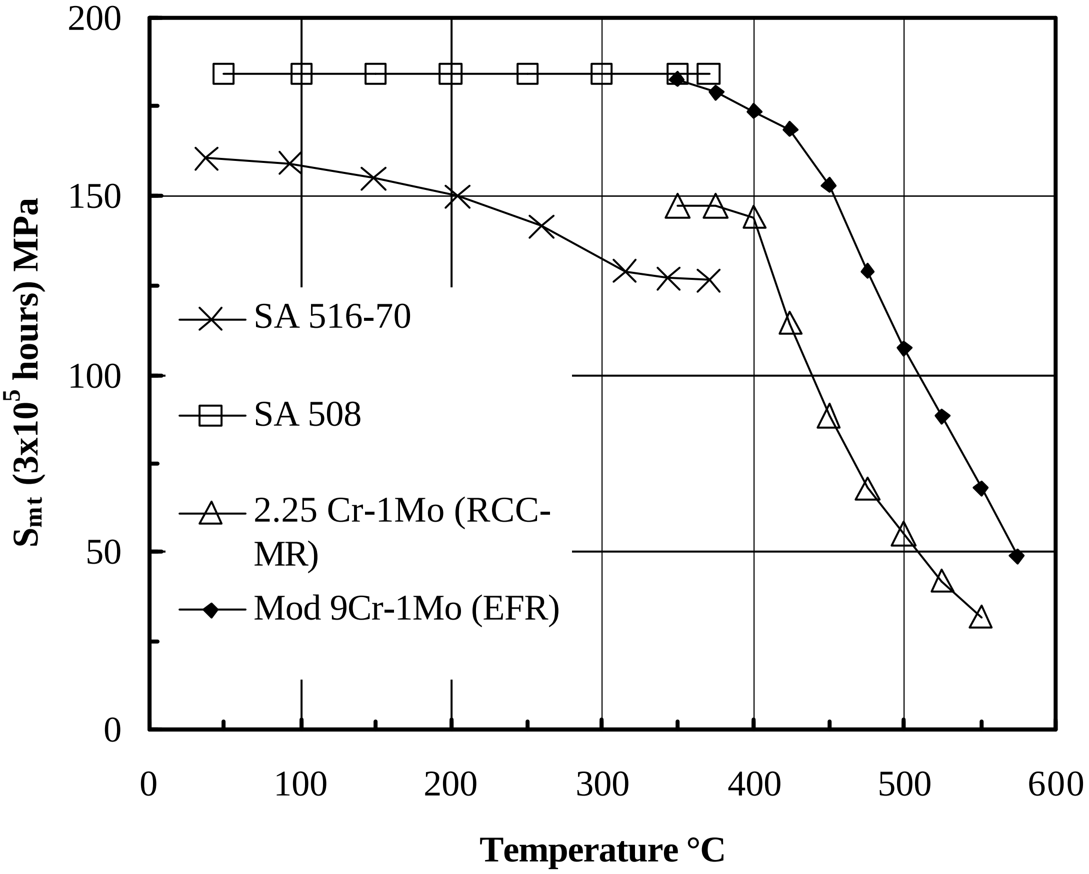


Figure 2. Effect of Irradiation on Ductile Brittle Transition Temperature (DBTT) of Cr-Mo Steels

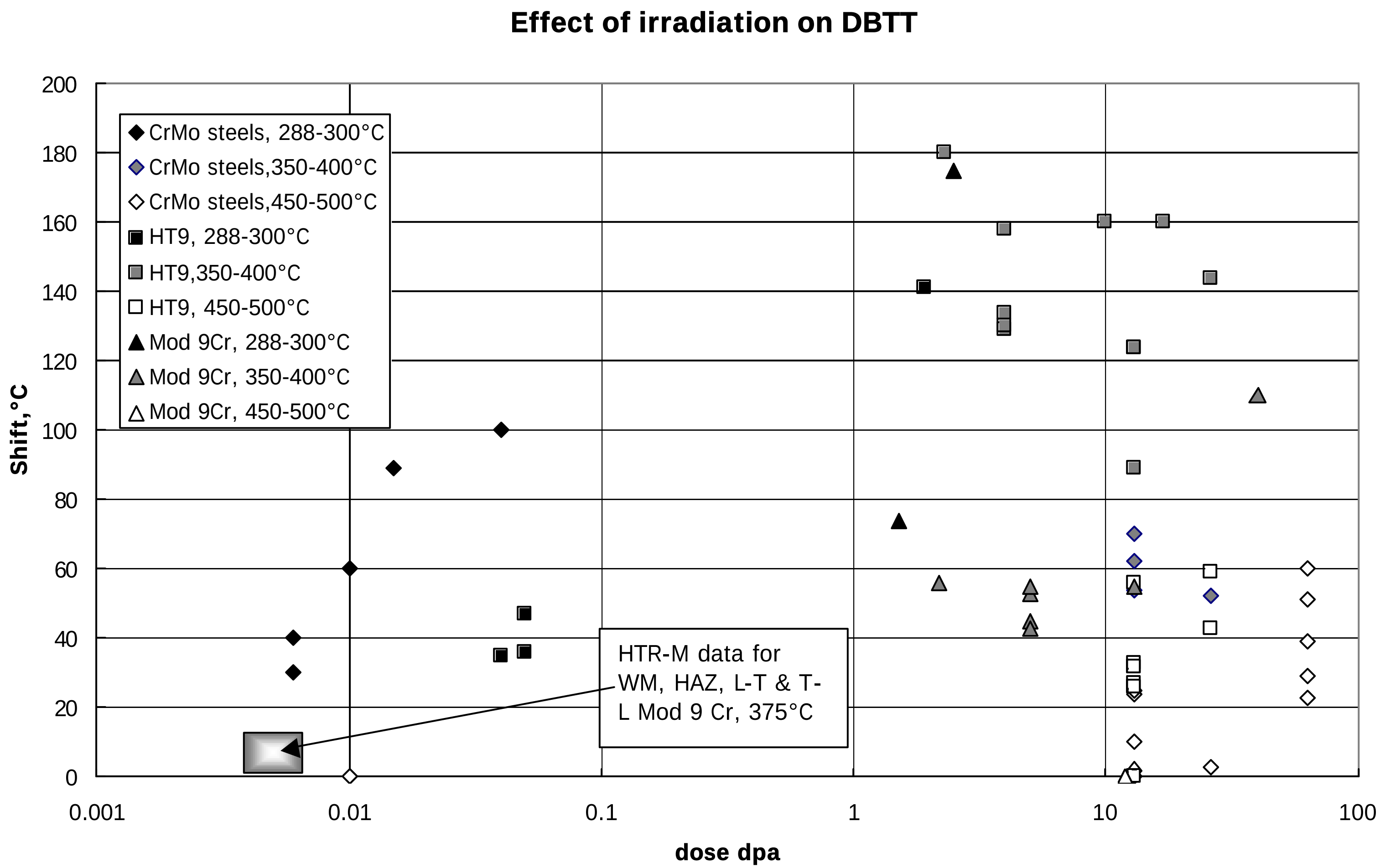
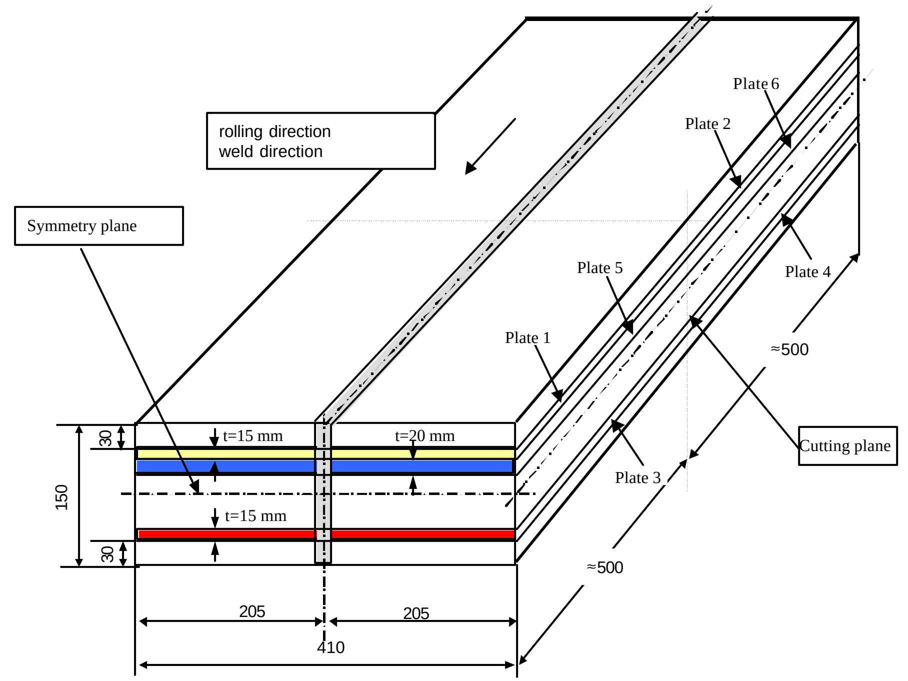
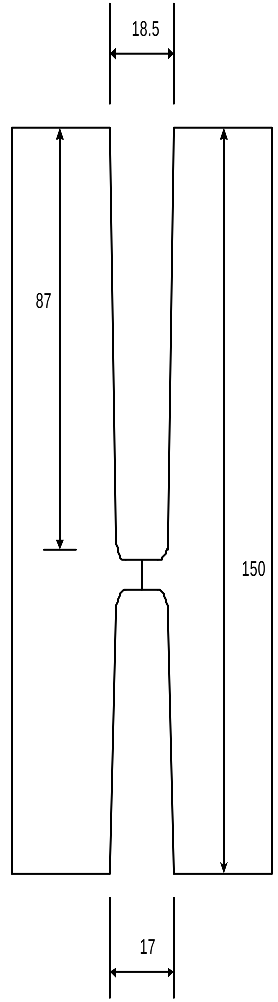


Figure 3 Vessel Welded Joint

Typical cutting plane of weld joint



Microcracks observed in welded joint



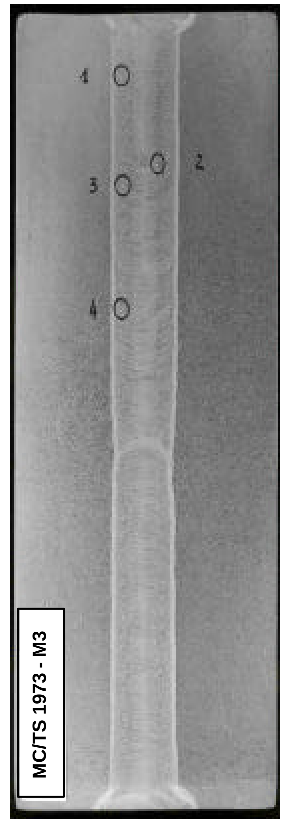


Figure 4 LYRA Test Rig and Loading

HFR Core

POOL SIDE FACILITY

WEST

	A	B	C	D	E	F	G	H		I
1	Be	Be	Be	Be	Be	Be	Be	Be	1	Be
2	F	F	F	D2	F	F2	F	H2	2	Be
3	F	F	C3	F	E3	F	G3	F	3	Be
4	F	CR	F	CR	F	CR	F	H4	4	Be
5	F	F	C5	F	E5	F	G5	F	5	Be
6	F	CR	F	CR	F	CR	F	H6	6	Be
7	F	F	C7	F	E7	F	G7	F	7	Be
8	F	F	F	D8	F	F8	F	H8	8	Be
9	Be	Be	Be	Be	Be	Be	Be	Be	9	Be

F Fuel element

CR Control rod

Be Reflectorelement

C5 Experiment position

Inserted in Pool side facility

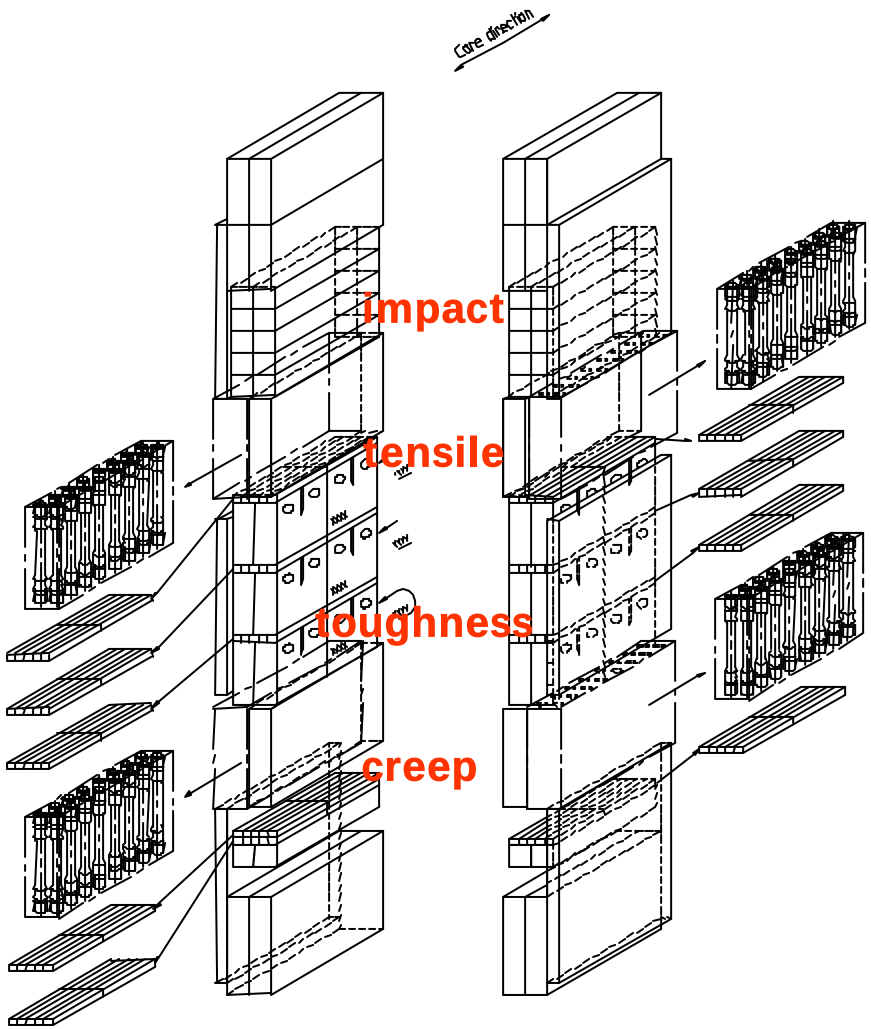


Figure 5 Stress Rupture Results for Modified 9 Cr weld at 550°C