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Advanced Nuclear Technology

Research Project ONR-RRR-088

Review of the iron-based materials applicable for the fuel and core of future Sodium Fast Reactors (SFR)

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Executive Summary

This research report investigates the available and developed materials for the next generation nuclear reactors and is intended to contribute to the development of knowledge on the area of advanced nuclear technologies (ANT) in the Office for Nuclear Regulation (ONR) by providing inspectors with a review of the publicly available information.

The operational loads and environmental conditions associated with these new nuclear technologies impose many material and fabrication challenges for which ONR needs to develop more detailed knowledge relating to reactor core designs in ANT.

This report investigates the iron-based materials that could be used for fuel cladding and core components operating under aggressive operating conditions. Sodium-cooled fast reactors (SFRs) are selected as an example for such conditions (erosive coolant, high temperature and high neutron flux), and wide range of operational experience (OPEX).

A set of criteria have been established and applied for identification of the gaps in the publicly available knowledge regarding the properties, applicability and availability of such materials.

The table below outlines the key findings of the study.

SFR Duct/ Cladding Material	***Key examples of the observations identified in this research report:**		
	Operational Experience (OPEX)	Material's Limitations	Industrial availability
Austenitic steels	▪ Dounreay Fast Reactor (DFR) and Prototype Fast Reactors (PFR)	▪ Good thermal creep resistance, ▪ High-temperature (500-600ºC) mechanical strength ▪ No ductile-to-brittle transition temperature ▪ Inherently susceptible to irradiation-induced swelling	▪ Established fabrication techniques in many countries
HT9 Ferritic-Martensitic Steel	▪ Leading candidate for ducts and fuel cladding in SFRs. ▪ EBR-II: test fuel cladding. ▪ FFTF: test ducts and fuel cladding.	▪ Swelling tolerable <208 dpa. ▪ Operated successfully between 350ºC-550ºC. ▪ Knowledge gap in low temperature embrittlement (loss of ductility) (<200ºC).	▪ No manufacture available worldwide. ▪ Defined in ASTM A771. ▪ No HT9 in ASME BPVC Code Section III Subsection NH – Class 1.
T91 Ferritic-Martensitic Steel (Mod 9Cr-1Mo)	▪ Near-term (>10 yrs) candidate. ▪ EBR-II, FFTF, Phénix, BOR-60: test specimens only.	▪ Swelling tolerable at <208 dpa. ▪ Max operating temperature 600ºC-650ºC. ▪ No OPEX as fuel cladding and ducts.	▪ Industrial scale manufacturing available. ▪ Defined in ASTM A213 A182 and A335. ▪ Produced today for coal power industry.
Oxide Dispersion Strengthened (ODS) Steel	▪ Long-term (>15-20 yrs) candidate. ▪ HIFR, BOR-60: test specimens only. ▪ BN-600: experimental fuel cladding since 2010.	▪ No OPEX as fuel cladding and ducts. ▪ Shown limited 'radiation resistance' properties. ▪ Unsatisfactory amount of data available to provide confidence in limitations.	▪ No worldwide manufacture available. ▪ Batch-to-batch variability. ▪ Heavy R&D on going.

* Further detailed findings are found in Appendix 12.3.

<u>Note</u>: The opinions of the author expressed in this report do not represent ONR or their regulatory opinion.

List of Abbreviations

AMR	Advanced Modular Reactor
ANT	Advanced Nuclear Technology (-ies)
ASTM	American Society for Testing and Materials
ASME	American Society of Mechanical Engineers
bcc	Body Centred Cubic
BEIS	UK Department for Business, Energy and Industrial Strategy
BPVC	Boiler and Pressure Vessel Code
BU	Burn Up
CEA	French Alternative Energies and Atomic Energy Commission
CRBR	Clinch River Breeder Reactor
DBA	Design Base Accident
dpa	Displacements per Atom
DFR	Dounreay Fast Reactor
DBTT	Ductile-to-Brittle Transition Temperature
EBR	Experimental Breeder Reactor
fcc	Face Centred Cubic
FFTF	Fast Flux Test Facility
FM	Ferritic/Martensitic
HFIR	High Flux Isotope Reactor
HTGR	High Temperature Gas-cooled Reactor
IAEA	International Atomic Energy Agency
ITER	International Thermonuclear Experimental Reactor
JAEA	Japanese Atomic Energy Agency
JSFR	Japanese Sodium-cooled Fast Reactor
ODS	Oxide Dispersion Strengthened
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PFR	Prototype Fast Reactor
PWR	Pressure Water Reactor
SFR	Sodium-cooled Fast Reactor

TABLE OF CONTENTS

List of Figures

<u>Note</u>: *Figures 2, 3, 8, 9, 15, 16, 17, 21 are reproduced from other publications under appropriately issued permissions by Elsevier and Copyright Clearance Center (TRIM: 2018/250508)*

List of Tables

1 Introduction

1. The Clean Growth Strategy policy paper [1] was released on 12 October 2017 by the Department of Business, Energy & Industrial Strategy (BEIS) which announced immediate investment into further development of the capability and capacity of the Office for Nuclear Regulation (ONR) to support and assess advanced nuclear technologies. This research project contributes to ONR's efforts to meet this objective.

2. One of the overarching themes in reviewing the sodium-cooled fast reactors (SFRs) is that their cladding and fuel assembly components experience intense neutron irradiation, high temperatures and corrosive and erosive coolants that exhaust the use of conventional materials (e.g. those used in light-water reactors). This creates the need for new nuclear materials to be developed for cladding and fuel assemblies [2, 3].

3. In addition, the trends to increase the demanding performance parameters and competitive fuel burn up (BU) – hence level of irradiation in other advanced nuclear technologies (ANT) – including SFR - have resulted in a design evolution that could outpace the accumulation of operational experience (OPEX).

4. High BU is desirable because it optimises resource use, reduces waste and improves economic viability of a nuclear power plant. Previous experimental SFR BU's were limited to the structural material's swelling and reduction of mechanical properties [2]. This limitation could be overcome by introducing new materials which exhibit lower swelling rates, operate at higher temperatures and maintain their mechanical properties when irradiated. These favourable properties could increase the BU in a commercial SFR design [4, 5].

5. The review is based on a top down approach using the following sequence:
    a. What is the SFR OPEX?
    b. What materials have been selected for fuel cladding and fuel assembly ducts?
    c. What are the challenges to these materials under SFR operational conditions?

6. In addition, a review of the key new materials proposed in new reactor concepts is carried out which identifies certain gaps in the publicly available knowledge. A view on the manufacturing capabilities and supply chain is also established.

7. The study was focused on the mechanical properties (tensile stress, impact, fracture toughness and creep) and swelling of unirradiated and irradiated samples of each material. It should be noted that corrosion of these materials is not investigated in this report. Detailed information on main corrosion challenges of steels by sodium is presented in the report released by CEA –France [6].

8. Section 2 of this report outlines the scope of this research project and the defined criteria for assessment of new materials. For each material investigated, the main findings are outlined in Sections 5.2, 6.3, 7.3 and 8.7. These sections discuss the review of the knowledge gaps related to four materials that could be used for SFR fuel assembly ducts and fuel cladding: austenitic stainless steels, HT9 ferritic/martensitic steel, T91 ferritic/martensitic steel and Oxide Dispersion Strengthened (ODS) steel.

2 Project Scope and Criteria

9. SFR's designs, materials, environmental conditions, components and operational envelopes have a spectrum of parameters. A set of criteria is outlined to assess new materials to determine their technological readiness for use in an SFR design.

10. This research project's scope includes:

- SFR technology (based on OPEX only).
- Fuel cladding materials for SFRs.
- Fuel assembly duct materials for SFRs.

11. This research project's scope is constrained by:

- The knowledge gaps identified within ONR as part of the terms of reference of this project.
- The author's knowledge and expertise in steel.
- The specified scope of the RRR-088 research project.

12. Six main criteria have been identified and they are: radiation damage limit, swelling, operational temperatures, low-temperature embrittlement, OPEX and manufacturing. Specifically, the duct and cladding materials are assessed against these criteria. These six criteria are based on reviewing various commercial and test SFR designs and should provide broad assessment of the material selection that could be applicable to any SFR reactor design [2, 7]. The criteria are:

1) Can the material survive irradiation (dose) beyond 100 dpa?

- Irradiation of the ducts and fuel cladding is expected to be above 100 dpa [8, 9]
- Is there sufficient knowledge of the material's mechanical properties (tensile stress, shift in ductile-to-brittle transition temperature (DBTT), fracture toughness and creep resistance) beyond this dose amount?

2) Is the radiation-induced swelling tolerable?

- Volumetric swelling can only be tolerable up to 5% [8, 10] over the fuel cladding and ducts lifetime. Beyond this percentage, the safety margins could decrease, and the risk could significantly increase.

3) Is there OPEX available for temperatures between 500 and 600ºC?

- This is based on of test facilities or on industrially operated SFRs [3, 11]

4) Is there OPEX available for the fuel ducts between 350-500ºC?

- This is based on of test facilities or on industrially operated SFRs [3, 11]

5) Is the material ductile below 200ºC?

- Susceptibility to low-temperature (0-200ºC) embrittlement is a consequence of using ferritic/martensitic (FM) steels [8, 10].
- FM steels exhibit relatively high DBTT [12].

6) Is the supply chain available to produce the selected material?

13. As outlined in the associated work plan [13], the use of new materials for fuel pellet hold-down springs in fuel rods and spacer grids in fuel assemblies is discussed together with other aspects of new materials application in advanced nuclear reactors.

3 Logic of the applied Methodology

14. The wealth of information regarding materials for SFR reactor designs, research from national laboratories, universities and industry is vast. To achieve the project objectives and proceed in investigating new materials within the time constraints, a logical method was developed. The method is based on a sequence of key questions:

1) What information has been published relating to the OPEX?

- What was the design, objectives and intent of the operated SFRs?
- Where can the source of information be located? For example, much of Experimental Breeder Reactors (EBRs) experience on materials has been published in the Journal of Nuclear Materials and the laboratory's website.

2) Considering the OPEX, what is the known safe operating envelope of the selected materials?

- What are the known limits of these materials?
- Are these limits under normal or accident conditions?
- Did the knowledge gained consider the synergistic effects (coolant compatibility, corrosion, irradiation damage, creep etc.)?
- Does the OPEX include laboratory experiments?
- What are the underlining mechanisms that cause implementation issues? For example, the use of austenitic steels in EBR-II produced structural integrity issues due to the unexpected significant swelling. This OPEX ultimately influenced the material selection for the Fast Flux Test Facility (FFTF).

3) What were the driving factors that changed the material selection?

- Is the structural integrity impacted with the continued use of these materials? For example, the inherent swelling rate of austenitic stainless steels is incompatible with long-term use under extensive radiation damage (>50 dpa).
- If the operating parameters were extrapolated, what is the confidence level? For example, HT9 steel's application is limited to approximately 600ºC. What is the risk of increasing the temperature beyond 600ºC?

4) What are the driving factors for new materials?

- Do these new materials allow for the operation in a new regime?
- Are the safety margins increased or decreased with using these new materials?

5) What evidence is used by the designers when selecting the materials?

6) Is there adequate capability and capacity to manufacture components in the selected materials?

4 Materials Selection for Ducts and Fuel Cladding of Previous and existing SFRs

15. This study begins with the available OPEX. The OPEX for SFRs is extensive compared to lead-cooled fast reactors, molten salt reactors and high temperature gas-cooled reactors (HTGRs) [2]. The information is based on public academic journals, national laboratory reports, International Atomic Energy Agency (IAEA) reports and industry reports. It should be noted that HTGRs material selection is not discussed in this report as ceramic materials are outside of the area for this study.

16. This section and the next discuss the history of reactor core (ducts and fuel cladding) materials in SFRs, the challenges faced, determination of the limitations, and modifications to materials to mitigate the challenges.

17. When analysing material science data, it is important to consider the experimental procedure carefully: does the experiment provide an environment where synergistic effects can take place (i.e. the material being exposed to stress, temperature, irradiation, coolant, thermal and stress fatigue etc.) compared to laboratory experiments that generally investigate degradation mechanisms or change in isolated material properties. It is important to distinguish between these studies.

18. It should be noted that the ducts and fuel cladding may experience cycles of stress relaxation, creep ductility, plastic strain to failure and stress-corrosion cracking. These factors are not considered in this report due to time constraints. However, these factors are important and should be considered in future studies.

19. There is clear evidence for the use of materials in SRFs from the 1960s-1980s; austenitic stainless steels were used however they all swell to the point that they could affect the structural integrity of the reactor core [11]. FM steels replaced austenitic stainless steels in SRFs from 1980s onwards to reduce the swelling. However, use of FM materials is potentially limited by their high temperature creep resistance and low-temperature embrittlement [12].

20. The general coolant (liquid sodium) inlet temperatures range is 250-400ºC and maximum temperature is between 650-700ºC [3]. The ducts and cladding components must withstand the continuous microstructural alteration from energetic neutrons colliding with atomic nuclei. This alteration could arise from displacements by neutrons and transmutation of elements producing helium (or hydrogen).

21. Section 5 discusses the use of austenitic stainless steels in SFRs, why they were chosen in the 1960s and the fundamental intrinsic undesirable property that arises from radiation damage. Section 6 discusses HT9 FM steel and identifies the knowledge gaps. Section 7 investigates an alternative T91 FM steel and identifies the knowledge gaps. Section 8 investigates the ODS steels that are currently developed as a potential material for future reactors. All four materials are assessed against the criteria outlined in Section 2.

22. Appendix 12.1 includes the definitions of material science terms that are used throughout this report.

5 Austenitic Stainless Steels (316, 316Ti, 15-15Ti, D9, AIM1)

23. Austenitic stainless steels were first selected as duct and fuel cladding based on their corrosion resistance, good thermal creep resistance, high-temperature (500-600ºC) mechanical strength, experienced fabrication techniques and no ductile-to-brittle transition temperature [2].

24. These steels were used in the Dounreay Fast Reactor (DFR) and Prototype Fast Reactors (PFR) [14], EBR-I [4] and EBR-II [4], Phénix [7] and Superphénix [7], JOYO [15] and Monju [15] SFRs (this is not a complete list). Additionally, these steels are extensively used within all of the Generation II/III(+) reactor cores [2]. The OPEX for these steels is extensive; hence they are seen as the benchmark for comparison.

25. The austenite name derives from the face-centred-cubic (fcc) crystal structure of iron. Steels at room temperature generally have the ferrite (body-centred-cubic (bcc)) crystal structure. Ferrite is known to undergo the DBTT and austenite does not. Austenite is metastable and will transform to ferrite when cooled below 750ºC. Certain alloying additions are used to stabilise the austenite phase at room temperature. These additions are nickel, manganese, silicon and carbon [16]. Nickel is chosen (for example, 316 austenitic stainless steel has 10-13 wt% Ni) to stabilise austenite, which also improves yield strength and oxidation resistance.

5.1 Swelling of Austenitic Stainless Steels

26. The major disadvantage to using austenitic stainless steels is that they are inherently susceptible to irradiation-induced swelling phenomenon. Swelling is a volume change mechanism.

27. Early results in the DFR [14] noticed that austenitic stainless steels were highly susceptible to radiation-induced swelling. This leads to swelling of the hexagonal ducts and bow that produce difficulties in removing and replacing them [17].

28. It was noticed in the PFR [14], EBR-II [4] and Phénix [18] that either modifications to these austenitic stainless steels or change to FM steels, which are highly resistant to radiation-induced swelling, were required.

29. Additions of titanium (Ti) to austenitic steels were found to be very effective in reducing the swelling behaviour. These Ti-stabilised alloys were used in Phénix and Superphénix (316Ti and 15-15Ti) however it was found that this modification only delayed the onset of swelling [3].

30. In all cases of austenitic stainless steels used in previous reactors, they undergo a similar process:

- Initial low-swelling transient regime. This region has been prolonged with the addition of Ti and cold-work.
- Significant swelling starts when the dose passes a threshold.
- Approximately 1% volume expansion per dpa after the threshold is passed [2].

This swelling process is outlined in Figure 1.

The swelling mechanism is further discussed in Appendix 12.2.

31. With the BU increase in SFRs, the irradiation damage levels are expected to reach >100 dpa [2, 19]. All austenitic stainless steels that receive this amount of damage will fail by extensive swelling (i.e. will likely rupture). This swelling can be seen in Figure 1. These steels are therefore unsuitable for the ducts and fuel cladding with reactor designs that increase BU beyond 50-100 dpa (the limit value is dependent on the austenitic steel used) [8].

32. It is important to note that the use of austenitic steels for cladding might halt the efficiency targets of a fast reactor [7] due to the BU restrictions.

33. It should be made clear that these steels do not limit power production. BN-600, a Russian commercial SFR in operation, has successfully generated power using austenitic stainless steel for the fuel cladding [5]. To further improve economics, an increase in BU is required. The next generation BN-1200 has been paused for a re-design to meet the economics of a water-cooled-water-moderated WWER [20] and part of this redesign is intended to increase the SFR BU.

34. It should be noted, even though it is outside the scope of this research project, that lead-based coolants dissociate nickel in steel alloys effectively rendering all austenitic stainless steels unsuitable for designs with lead-based coolants [21]. This fact indicates the need for a new material that does not contain significant amount of nickel (for example, FM steels).

35. The swelling behaviour and magnitude of austenitic stainless steels was unexpected. This point highlights the fact that full scale testing for synergistic effects must be considered in any new material proposed in an SFR,.

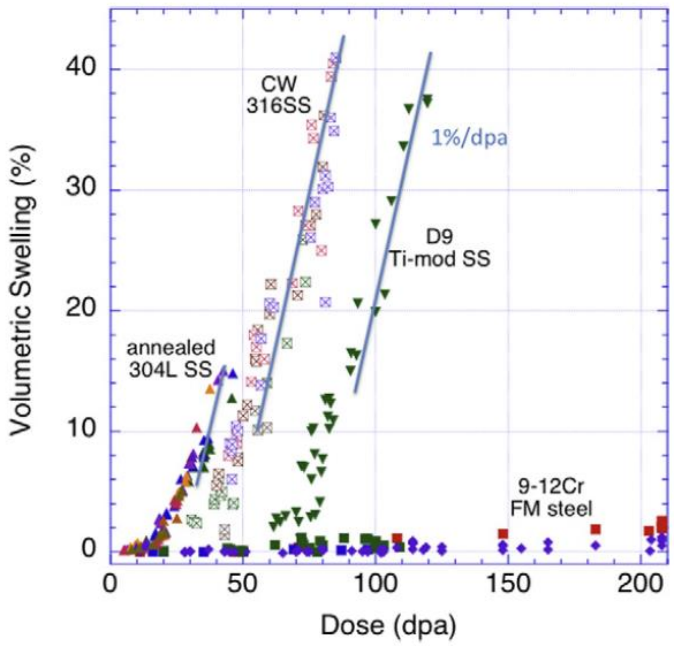

Figure 1: Comparison of the volumetric swelling behaviour of stainless steels (304L SS, CW 316SS, D9 Ti-mod SS; CW (cold work), SS (stainless steel)) and FM 9-12Cr steels following fast neutron irradiation between 400-550ºC temper ature in Phénix. This is a good example of the severity austenitic stainless steel's radiation-induced swelling is compared to FM steel's swelling. Figure reproduced from [2].

5.2 Assessment of Austenitic Stainless Steels

36. The criteria outlined in Section 2 are used to assess the suitability of using austenitic stainless steels in a large scale SFR reactor. The outcomes are listed in Table 1.

37. From the findings it is clear that the swelling of austenitic stainless steels is unacceptable beyond 100 dpa. It is expected for commercial SFRs that they will reach 100+ dpa for the fuel cladding [2, 8]. A different type of steel, which is more resistant to swelling, must be selected instead.

38. There are minimal knowledge gaps if new SFR operational envelopes are similar to the previous test reactors. For a commercial SFR, the operational envelopes appear to be greater (>100 dpa damage to the fuel cladding) hence the knowledge gaps increase.

39. Austenitic stainless steels are intrinsically susceptible to radiation-induced swelling. There have been attempts to mitigate (by adding Ti or cold work) however these methods have only prolonged the swelling. Therefore, the only apparent option for a commercial reactor is to either change the material or reduce the operational envelope.

Table 1 Assessment for the suitability of using austenitic stainless steels as the duct and fuel cladding for the SFR reactor core against the criteria outlined in Section 2.

Criteria	Assessment
Can the material survive >100 dpa?	Very limited – see swelling
Is the swelling tolerable?	No – ALL austenitic stainless steels swell beyond 5% within the dose region of 50-100 dpa. This is unacceptable and unsuitable for large scale SFR. Mechanical properties (tensile stress, impact, fracture toughness and creep) are overshadowed by the intolerable use of these steels beyond the swelling limit.
Is there OPEX available for the fuel cladding between 500-600ºC?	Yes – significant OPEX within this temperature region.
Is there OPEX available for the ducts between 350-500ºC?	Yes – significant OPEX within this temperature region.
Is the material ductile below 200ºC?	Yes – OPEX within this temperature region. It should be noted that these steels do not exhibit a DBTT.
Is the supply chain available?	Yes

6 HT9 Ferritic/Martensitic Steel (12Cr-1MoVW)

40. HT9 FM steel (also known as 12Cr-1MoVW) was designed by Sandvik in Sweden and used as ducts and fuel cladding in FFTF, USA, as it showed great promise on resistance to swelling and excellent compatibility with liquid sodium at operating temperatures. FFTF was the first reactor where HT9 was significantly used to investigate the longevity, coolant compatibility and performance relevant to use in future commercial SFR reactors [22].

41. In general terms, FM steels have high thermal conductivity, good high-temperature creep resistance, low thermal expansion coefficient and low swelling rate. These properties allow FM steels to be considered for fuel cladding and ducts in SFRs [17].

42. The composition limits of HT9 are defined in American Society for Testing and Materials (ASTM) A771 [62] for seamless austenitic and martensitic stainless steel tubing for SFRs. The composition is listed in Table 3.

Table 2 The upper and lower bounds of HT9 alloy composition defined in ASTM A771 specification for stainless steel tubing.

	Composition (wt %)										
Steel	C	Cr	Ni	Mo	W	V	Mn	Si	P	S	Nb
HT9 (upper bound)	0.23	12.5	0.8	1.2	0.6	0.35	0.7	0.3	0.04	0.01	0.05
HT9 (lower bound)	0.17	11.0	0.3	0.8	0.4	0.25	0.4	0.2	0.04	0.01	0.05

Table 3: The upper and lower bounds of HT9 alloy composition defined in ASTM A771 [62] specification for stainless steel tubing.

6.1 Operational Experience

43. The last SFR to be constructed in USA was FFTF which operated successfully from 1982 to 1992. It was constructed on the Department of Energy's Hanford Site, Washington State. It consisted of a 400 MWth 3-loop reactor with oxide fuel in two enrichment zones. The inlet temperature was 360ºC and outlet temperature was 527ºC. The reactor aims were to provide extensive capability for in-core irradiation testing with independent instrumentation for specimens. This reactor has influenced the future designs of SFRs [23].

44. It is important to note that most irradiations of materials carried out by universities are based on simple free-standing specimens under well-defined and near-constant conditions. Fuel assemblies are complex in shape, have a profile of conditions across the component and are subject to various operational stresses.

45. FFTF's ACO-3 hexagonal duct, fuel cladding and wrapping wire (a wire that separates the fuel rods) were constructed out of HT9 steel. The irradiation of this assembly occurred over 6 years, giving quality information on the swelling resistance and properties of the material in industrial operation environment. Temperatures and doses ranged over a profile as a function of axial duct position.

46. An example of a good experiment investigating the degradation of materials was conducted by Sencer et al. [22]. This ACO-3 duct was deconstructed and tested, which provided a high quality set of mechanical properties across a range of conditions. The temperature and dose profiles of ACO-3 are shown in Figure 2. The duct was in contact with flowing liquid sodium throughout its operation. The temperature varied ±10ºC over the duct lifetime.

47. Sencer et al. concluded that the swelling was 0.3 % irradiated over 6 years at 443ºC reaching 155 dpa. This study has shown that HT9 steel's swelling resistance is excellent. It should be noted that the duct temperature was less than that of the fuel cladding.

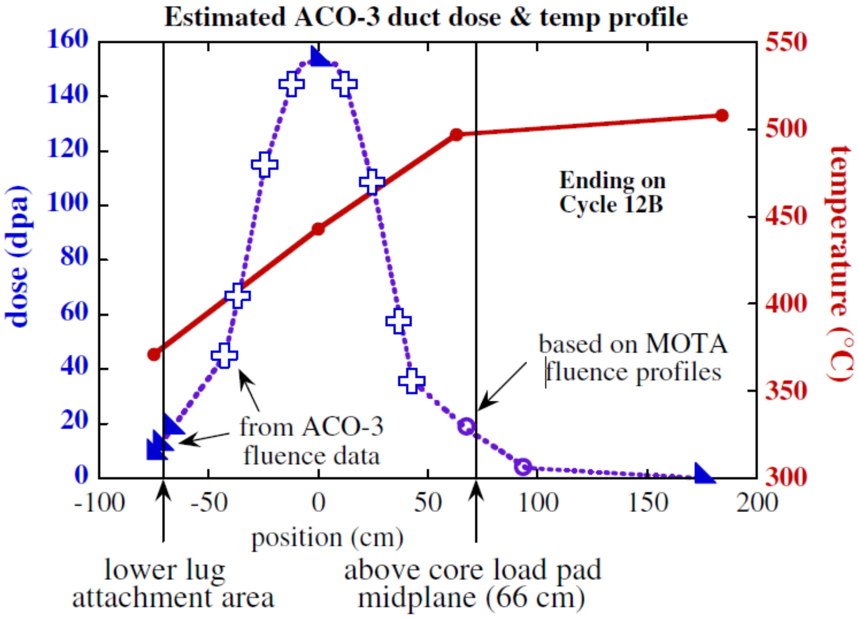

Figure 2: Irradiating conditions across the ACO-3 duct - reproduced from [22]. This is an example of high quality data applicable for investigating the effects of irradiation damage on HT9 steel. It provides a large data set of HT9 steel irradiated across a temperature and dose profile within an operating SFR.

48. Another important study conducted at FFTF was using helium-pressurised tubes of HT9 irradiated to 208 dpa at 400ºC [24]. Pressurised helium was used to simulate fission gas pressure on the inner surface. It should be noted that the reactor's thermal output was lowered from 400MWth to 280MWth to reduce the temperature of the alloys. At 400 MWth, the temperature would have been beyond the high temperature limit (550-600ºC). This reduction in power indicates that the steel is unable to operate at high temperatures (550-600ºC).

49. A lesson learned from the FFTF reactor was that significant risk is involved in completing a design without using sufficient material data is [23]. FFTF designed reflector assemblies that penetrated stainless steel to provide cooling tubes and reflect neutrons to be constructed out of Inconel-600. During one of the refuelling outages, difficulties were discovered when removing the reflector assembly because the amount of swelling was higher than expected. This is a very important lesson which indicates that using materials that have previous OPEX is paramount.

50. To reaffirm the suitability of using FM steels as reactor core material, the Russian SFRs have used EP-450 (Type 13Cr2MoNbVB) FM steel (similar to HT9 steel) as the duct material in both BN-350 and BN-600 reactors [25].

51. A study of these ducts [26] outlined favourable swelling resistance and acceptable performance (tensile strength, shift in DBTT) at high irradiation doses in the 400-550ºC temperature range.

52. At medium operating temperatures (380-400ºC), the maximal damage occurred at 10-40 dpa indicates that EP-450 FM steel is susceptible to low temperature embrittlement. This is caused by the increase in DBTT and the reduction in the upper-shelf energy.

53. The Russian EP-450 steel would unlikely be chosen for western designed SFRs. It is cited here to indicate and provide an example of the successful FM steel use. Notably, this steel is adopted as the material for the fuel-assembly duct in BN-800 [25].

54. An extensive dataset is compiled from various sources and provides an overview of the known operational window for HT9 FM steel. This is shown in Figure 3. It is clear that there is a lack of data for operating below 200ºC and beyond 147 dpa.

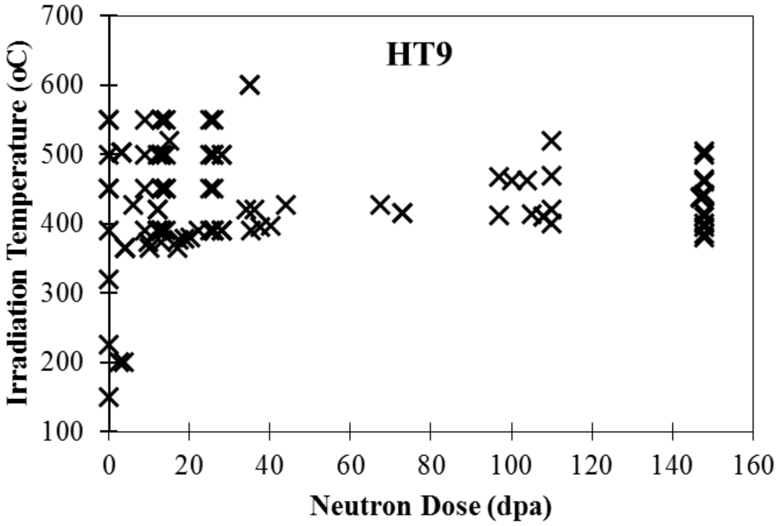

Figure 3: OPEX available for HT9 steel. Mechanical data has been acquired for each data point. It is important to note that no data exists beyond 208-210 dpa (swelling and creep only), 147 dpa for mechanical properties and below 200ºC. Un-irradiated conditions are included at 0 dpa for comparison. Data collected from [10, 24, 27, 28, 29, 30]. The raw data is presented in Appendix 12.3.

6.1.1 Tensile Stress Data

55. Key studies in understanding the tensile yield stress (tensile strength) under irradiating conditions were carried out during the operation of EBR-II and FFTF [31, 32,27]. The main results are compiled in Figure 4. Reports from non-fast neutron spectrums have not been included [33] because major differences in neutron energy spectrums (e.g. fast versus thermal) produce different amounts of damage.

56. There is currently no tensile stress data beyond 67.5 dpa and for low temperature irradiation in an SFR environment (<373ºC).

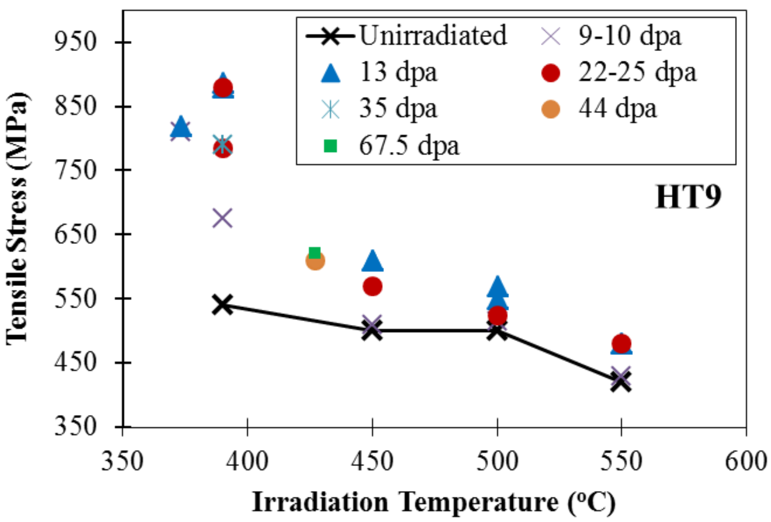

Figure 4 Tensile yield stress (MPa) of irradiated HT9 steel in both EBR-II and FFTF SRF reactors [27, 31, 32]. There is no data below 373ºC. The raw data is presented in Appendix 12.3.

6.1.2 Impact Data (DBTT)

57. The DBTT shifts were compiled for HT9 steel irradiated data from operational experience and are shown in Figure 5.

58. The results indicate that HT9 steel is susceptible to low temperature radiation embrittlement (<200ºC) because there is a high shift in DBTT and increase in tensile stress. As the irradiation temperature increases, the embrittlement (shift in DBTT) decreases. This is due to increase in the rate of radiation defects recombination under higher temperature.

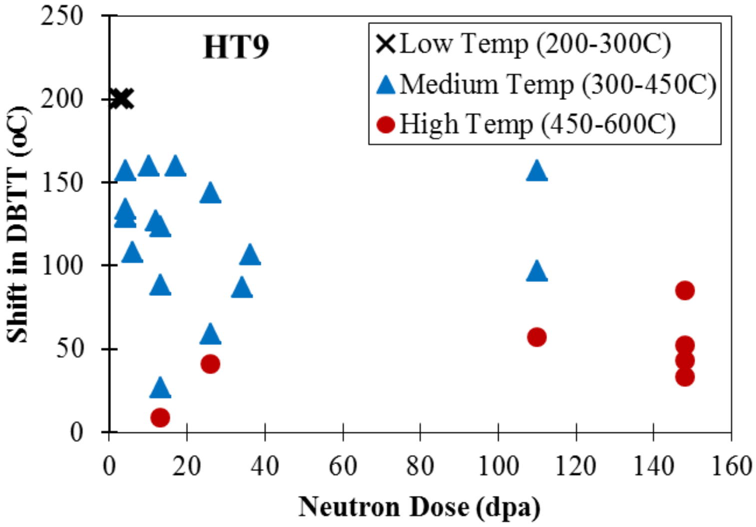

Figure 5 All known DBTT data available for HT9 steel that has been in operation. It is important to note that no data exists beyond 147 dpa and below 200ºC. Reproduced from [28] and from [30] – for 147 dpa. The raw data is presented in Appendix 12.3.

6.1.3 Fracture Toughness

59. Fracture toughness experiments were conducted on unirradiated and irradiated specimens in EBR-II and FFTF reactors. A compilation of the plastic-elastic J-integral values are presented in Figure 6. No OPEX on fracture toughness is currently known beyond 147 dpa.

60. A review by Los Alamos National Laboratory in 2012 [10] concluded that the fracture toughness of HT9 steel is sufficient for fast reactor irradiation of order up to 100 dpa.

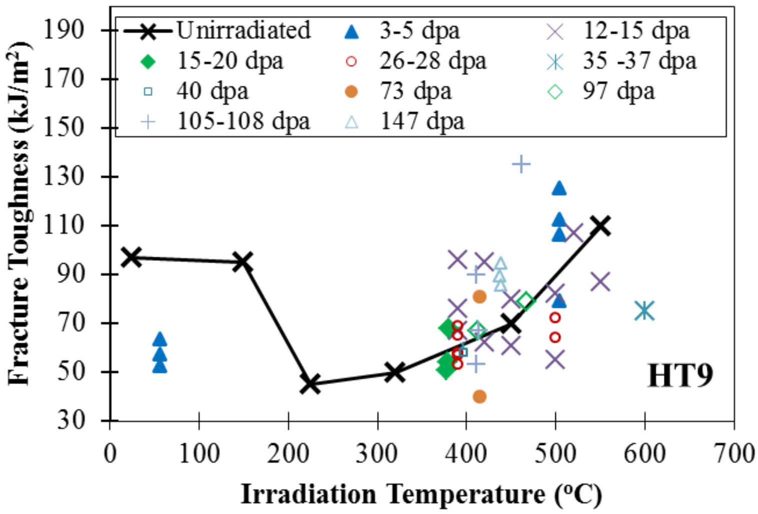

Figure 6 HT9 Fracture toughness values (J-integral) of unirradiated (line) and irradiated in EBR-II and FFTF reactors. This is a compilation of the data from [34], [35] and [36]. The raw data is presented in Appendix 12.3.

6.1.4 Creep and Swelling

61. Experiments on HT9 steel were conducted in pressurised helium tubes to simulate hoop stresses seen in operating fuel cladding tubes. These helium-pressurised tubes were inserted into FFTF at 400ºC and received a total dose of 208 dpa [24].

62. Creep dimensional strains for HT9 steel are shown in Figure 7 at different levels of hoop stress and dose received.

63. Swelling was determined by measuring the density changes of the steel. After irradiated to 208 dpa at 400ºC in FFTF, the swelling of HT9 was reported to be 1-2.5% volumetric swelling. This is shown in Figure 8.

64. It should be noted that the amount of swelling observed in HT9 is significantly less than in austenitic stainless steels (see Section 5.1). This feature of FM steels is one of the main advantages of using them in a high radiation dose environment.

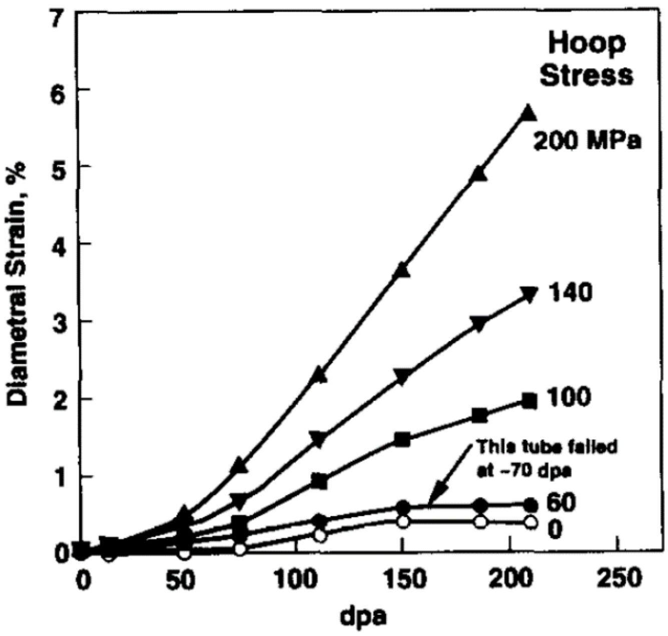

Figure 7 HT9 steel irradiated in pressurised tubes at 400ºC to 208 dpa in FFTF [24]. The dimensional strains (diametric strain %) were measured to determine the level of creep observed at different hoop stresses. Sodium was detected within the 'failed tube at 70 dpa'.

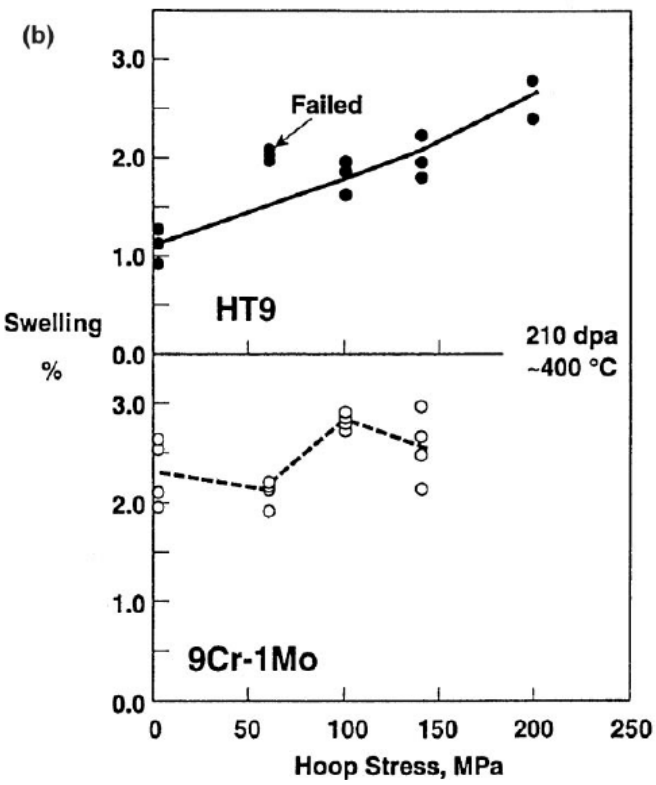

Figure 8 HT9 steel (top) irradiated to 208 dpa at 400ºC in FFTF swelling as a function of Hoop stress (MPa) that was simulated by using pressurised helium during irradiation [24]. T91 steel (bottom, 9Cr-1Mo) is discussed in Section 7.1.4.

6.2 Identification of Knowledge Gaps

65. The literature review of this project has outlined clear knowledge gaps with using HT9 steel for the ducts and fuel cladding in commercial SFRs:

- Majority of the research conducted on HT9 steel has been in FFTF and EBR-II reactor where BU (irradiation) was low. For a commercial reactor, the BU would likely increase [8] . However, knowledge on how the steel performs at such BU levels is currently missing.
- There has not been a study that explicitly investigates the synergistic effects.
- American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III, Division 5 – Class 1 Components in Elevated Temperature Service - does not list HT9 steel as a suitable material hence it needs to be adopted by ASME before licensing is possible in the USA.
- Long term creep performance for a 40/60-year design life time has not been demonstrated. (EP-450 is the only steel that has been used for a long term - in the BN-600 SFR.)
- Majority of the high dpa (>100 dpa) mechanical properties data have been derived from a duct in FFTF. The temperature of the duct was lower than the fuel cladding temperature.
- The knowledge gap on the mechanical properties of irradiated HT9 below 200ºC is significant as the shift in DBTT and increase in tensile strength indicates that low temperature embrittlement could cause the steel to fail during over-cooling faults.
- Knowledge of the material properties (tensile strength, impact, fracture toughness) of irradiated HT9 steel beyond 147 dpa is missing.
- This review has not identified a manufacturer producing HT9 steel [10].
- Swelling and creep properties are unknown beyond 208 dpa at 400ºC.

6.3 Assessment of HT9 FM Steel

66. The criteria outlined in Section 2 are used to assess the suitability of using HT9 in a commercial scale SFR. The outcomes are listed in Table 4.

67. From the findings, it is clear that the swelling of HT9 steel is acceptable up to 210 dpa. The fuel cladding of commercial SFRs is expected to reach >100 dpa [2, 8]. The maximum operating temperature is between 550-600ºC [8]. However, HT9 steel is susceptible to low-temperature embrittlement.

68. FFTF studies on HT9 have produced a dose-temperature parameters envelope with no apparent knowledge gaps which could be convenient for the design of future SFRs [8]. The reviewed studies have not identified an intrinsic material property that would inhibit HT9 to be used in this environment (compared to significant swelling in austenitic stainless steels).

69. However the application of these limits would significantly affect the economic viability of a power-generating plant [8].

70. It then appears that HT9 could be unsuitable to use as fuel cladding and/or duct material for a power generating plant that wishes to increase BU (i.e. increase the radiation damage limit) and possibly the operational temperature.

Table 4: Assessment for the suitability of using HT9 FM steel as fuel cladding and ducts for the SFR reactor core against the criteria outlined in Section 2.

Criteria	Assessment
Can the material survive >100 dpa?	Limited knowledge – shift in DBTT up to 147 dpa, tensile stress up to 67.5 dpa, fracture toughness up to 147 dpa, creep dimensional strain up to 208 dpa.
Is the swelling tolerable?	Yes – swelling of HT9 ranges between 1-2.5% to 210 dpa at 400ºC, over a range of hoop stresses (0-200 MPa)
Is there OPEX available for the fuel cladding between 500-600ºC?	Yes – significant OPEX within this temperature region up to 550ºC (fracture toughness, shift in DBTT, tensile stress). Very limited from 550ºC to 600ºC (only a few experiments that measured fracture toughness to 35 dpa).
Is there OPEX available for the ducts between 350-500ºC?	Yes – significant OPEX within this temperature region up to 500ºC (fracture toughness, shift in DBTT, tensile stress).
Is the material ductile below 200ºC?	Limited knowledge – Tensile (yield) stress and DBTT show significant amount of embrittlement at these temperatures from a low dose (9-35 dpa).
Is the supply chain available?	No

7 T91 Ferritic/Martensitic Steel (Mod 9Cr-1Mo, P91, F91)

71. HT9 FM steel is limited by its high temperature strength and low temperature embrittlement [8]. To achieve higher BU and temperature, a new material should be selected [12, 8].

72. T91 FM steel is chosen as it has similar properties to HT9 FM steel however can operate at a higher temperature up to 650-700ºC with the same creep lifetime [12]. Vanadium nitrides, molybdenum carbide and martensitic microstructure enable T91 FM steel to operate at a higher temperature under stress (compared to HT9 steel). The increased resistance to low temperature embrittlement is achieved by reduction of Cr addition from 12% in HT9 to 9% in T91. More information on T91 is given in ref [12].

73. Grade 91 steel (also known as T91, P91, F91, Mod 9Cr-1Mo and as 9Cr-1Mo) is listed in the ASTM A213 (tubing), A182 (forgings) and A335 (piping) standards.

74. T91 is the name generally used within literature because the application is considered to be the cladding (i.e. tubing) form. The composition bounds are found in Table 5.

Table 5 The upper and lower bounds of T91 steel composition as defined in the ASTM A213 tubing specification.

	Composition (wt %)									
Steel	C	Cr	Ni	Mo	V	Mn	Si	P	S	Nb
T91 (upper bound)	0.14	9.5	0.4	1.05	0.25	0.6	0.5	0.02	0.01	0.6
T91 (lower bound)	0.07	8.0	0.4	0.85	0.18	0.4	0.2	0.02	0.01	0.1

75. Originally T91 was designed for the Clinch River Breeder Reactor's (CRBR) steam-generator. After the Fast Breeder Reactor program was closed in 1990s, T91 irradiation studies were carried out under the US Fusion Materials Program.

76. The main driver for using T91 FM steel for the steam generator tubes was its high thermal conductivity. CEA are considering T91 FM steel for future steam generators [7]. However they outline a fundamental issue - FM steels strain softening under fatigue cycling whereas austenitic steels strengthen under fatigue cycling [7].

77. T91 steel already has ASME Code Section III Subsection NH and ASTM manufacturing qualifications. The manufacturing capability has already been established because the steel is used in the coal power industry.

78. EM10 steel designed in France has similar composition and manufacturing route to T91 steel. EM10 has been used as duct material in Phénix and Superphénix SRFs [7]. The high resistance to swelling was the key reason for the change from 15-15Ti austenitic stainless steel to EM10 in these projects.

79. EM10 was the last improvement to the Phénix duct material [7] and is cited as an example of how the duct material improvements can lead to BU increase.

7.1 Operational Experience

80. There has been no operational use of T91 FM steel as fuel cladding or ducts in any SFR reactors.

81. Majority of the T91 irradiation has occurred in BOR-60 [37], FFTF [38, 24] and EBR-II [24] as test specimens. The steel has shown to resist swelling and embrittlement at medium temperature (450-500ºC) and demonstrated high temperature creep resistance (650ºC). However, there is little information how irradiation and coolant effects the mechanical properties over a long period of time (years) [8].

82. Japanese Sodium-cooled Fast Reactor (JSFR) program are studying long term mechanical properties (tensile stress and creep resistance) over 100,000 h (~11 years) of T91 to predict the performance for 60-year design lifetime [39] The study attempts to use the Larson-Miller parameter to extrapolate to a 60-year lifetime. The batch-to-batch variability of T91 has been incorporated into the models. It should be noted that these experiments were conducted on unirradiated specimens.

83. Only T91 steel test specimens irradiated in SFRs have been investigated over a range of neutron flux and temperature. Pressurised helium-filled T91 FM steel tubing was used to simulate hoop stresses in FFTF [24].

84. The majority of T91 irradiation data is shown in Figure 9. This figure outlines the main irradiation experiments on T91 within FFTF, BOR-60, Phénix and EBR-II reactors as test specimens. The synergistic effects from the operational environment are not considered.

85. Figure 9 indicates that the OPEX is limited to up to 105 dpa and between 50-550ºC.

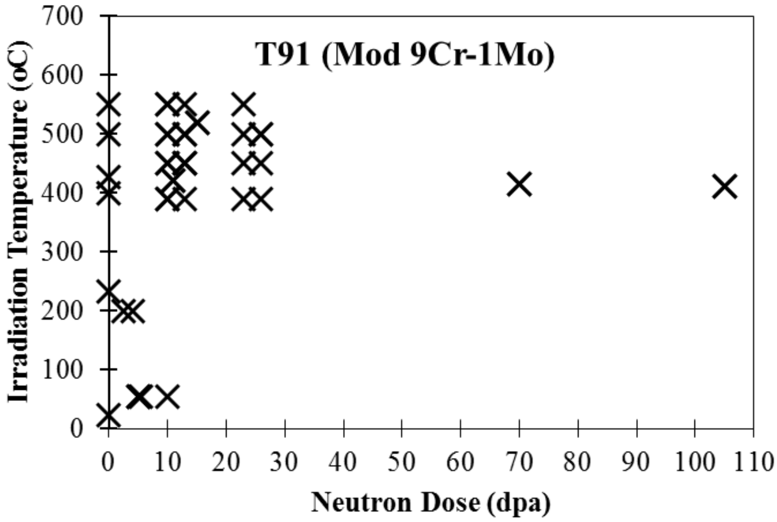

Figure 9 Compilation from various sources [29, 31, 35, 40, 41, 42] of irradiation experiments in SRFs (FFTF, EBR-II, BOR-60 and Phénix) of T91 steel test specimens. Unirradiated (0 dpa) test data is included for comparison. The raw data is presented in Appendix 12.3.

7.1.1 Tensile Stress

86. Tensile specimens were inserted to EBR-II and irradiated at 390, 450, 500 and 550ºC between 10 and 12 dpa. The tensile testing was conducted at the irradiated temperature quoted. The yield stress as a function of irradiation dose is shown in Figure 10.

87. Outside SFR technology and the scope of this research project, lead-cooled fast reactors have proposed using T91 as fuel cladding [21]. Belgian Nuclear Research Centre is designing an accelerator driven system that proposes to use T91 steel as the structural material with a lead-bismuth eutectic coolant for a subcritical core. Test T91 samples submerged in Pb-Bi eutectic were irradiated in BR2 test reactor to simulate reactor type conditions.

88. T91 was irradiated to 1.5 dpa between 460 and 490ºC [21] in Pb-Bi eutectic coolant. Tensile strength was softened by 50-100 MPa within the eutectic mixture when under irradiation conditions. No irradiation hardening was observed to 1.5 dpa. This is one of the few studies investigating T91 in a lead-cooled environment.

89. The lead-cooled study on T91 is an example of how this steel could be used for other technologies than just liquid sodium-based reactors.

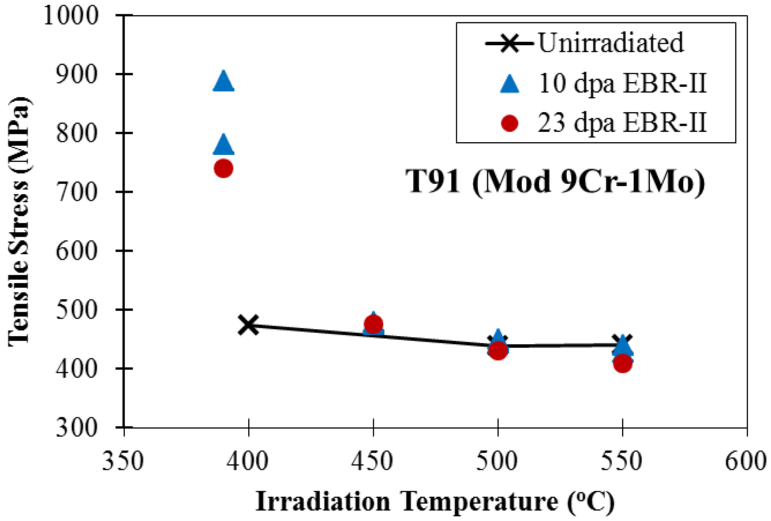

Figure 10 Tensile yield stress (MPa) T91 steel irradiated as test specimens in FFTF and EBR-II reactors. The data is a compilation of [31] and [41]. The raw data is presented in Appendix 12.3.

7.1.2 Impact Data (DBTT)

90. Figure 11 displays the shift in DBTT. In comparison to HT9, T91 appears less prone to low temperature embrittlement. This is due to the higher 12 wt% Cr in HT9 forms alpha-prime (α') precipitation (a high Cr rich bcc secondary phase) under irradiation.

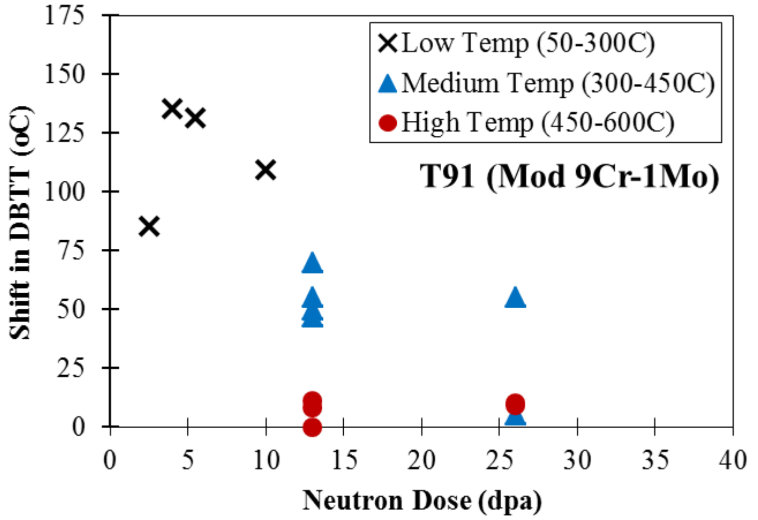

Figure 11 Compilation from various sources [29, 40, 42] of the shift in DBTT in T91 steel. The raw data is presented in Appendix 12.3.

7.1.3 Fracture Toughness

91. Fracture toughness experiments were conducted on unirradiated and irradiated specimens in FFTF. A compilation of the plastic-elastic J-integral values is presented in Figure 12.

92. There is limited fracture toughness data on irradiated T91 specimens.

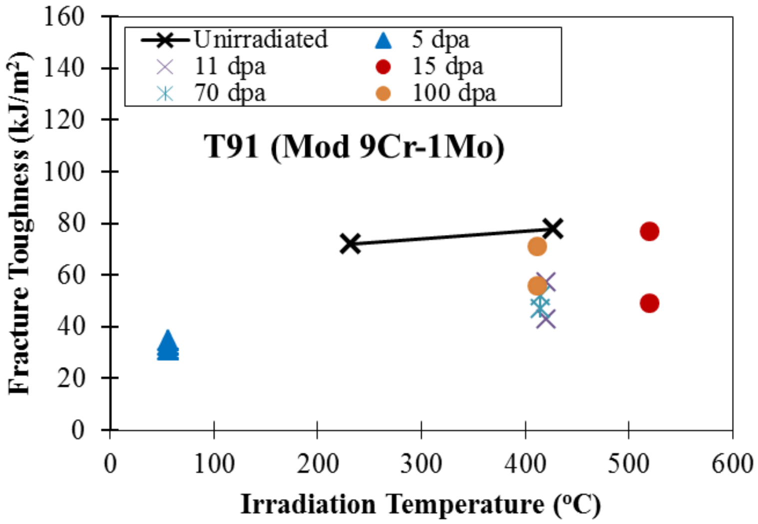

Figure 12 Fracture toughness (J-integral) values for unirradiated and irradiated T91 specimens in FFTF. This is a compilation of results from [35]. The raw data is presented in Appendix 12.3.

7.1.4 Creep and Swelling

93. Experiments on T91 steel was conducted in pressurised helium tubes to simulate hoop stresses seen in operating cladding tubes. These pressurised tubes were inserted into FFTF at 400ºC and received a total dose of 208 dpa [24].

94. Creep dimensional strains for T91 are shown in Figure 13 at different levels of hoop stress and dose received.

95. Swelling was determined by measuring the density changes of the steel. After irradiated to 208 dpa at 400ºC in FFTF, the swelling of T91 was reported to be 1-2.5 volumetric %. This is shown in Figure 8.

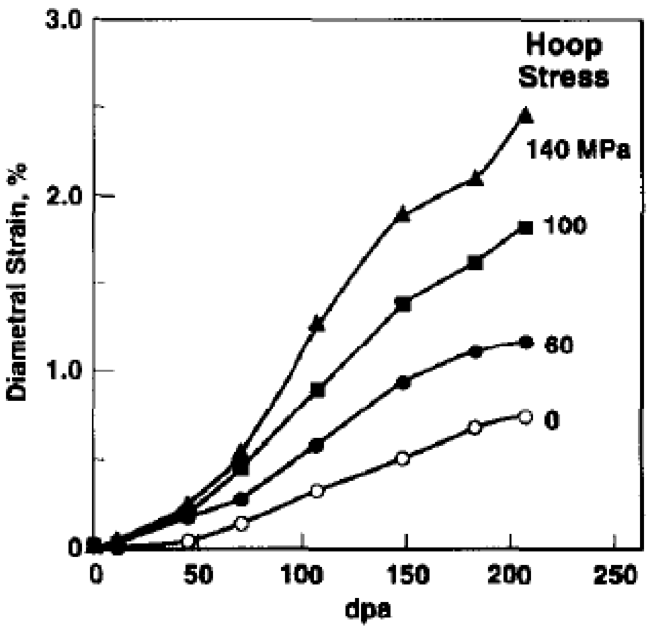

Figure 13 T91 steel irradiated in pressurised tubes at 400ºC to 208 dpa in FFTF [24]. The dimensional strains were measured to determine the level of creep observed at different hoop stresses.

7.2 Identification of Knowledge Gaps

96. The major knowledge gaps identified are as follows:

- OPEX with T91 steel is limited. No component constructed with T91 steel has been in operation in any reactor. Only test specimens have been examined in BOR-60, EBR-II, Phénix and FFTF. There is a gap in knowledge of how the material performs as a fully operational component. Further extensive tests using T91 as a duct or fuel cladding are required.
- Minimal amount of mechanical properties (tensile stress and impact data) are known for low-temperature radiation embrittlement. Beyond 23 dpa and 550ºC the mechanical properties are currently unknown.
- Fracture toughness is unknown beyond 105 dpa at high temperatures (>400ºC).
- There is no long term data for a lifetime exceeding 10 years.
- No high dose with high temperature swelling data exists for T91. The current data is limited to 208 dpa at 400ºC.
- Minimal data is available for understanding how T91 steel behaves under irradiation.
- No synergistic study has been conducted on T91.

7.3 Assessment of T91 FM Against the Criteria

97. The criteria outlined in Section 2 are used to assess the suitability of using T91 in a large scale SFR reactor. The outcomes are listed in Table 6. The swelling of T91 FM steel is acceptable up to 210 dpa at 400ºC. It is expected for commercial SFRs that they will reach 100+ dpa for the fuel cladding [2, 8]. Unirradiated T91 FM steel can operate under stress to 650-700ºC. However, no irradiation studies have been conducted at these temperatures to understand if the material is suitable [12].

98. Overall, the knowledge gap is large when assessed against the criteria of this project. There is no OPEX of T91 FM steel as cladding or duct components in a SFR to date.

Table 6 Assessment for the suitability of using T91 steel as fuel cladding and ducts for the SFR reactor core against the criteria outlined in Section 2.

Criteria	Assessment
Can the material survive >100 dpa?	Very limited knowledge – shift in DBTT up to 26 dpa, tensile stress up to 23 dpa, fracture toughness up to 100 dpa, creep dimensional strain up to 208 dpa.
Is the swelling tolerable?	Yes – swelling of T91 ranges between 2-3% to 210 dpa at 400ºC - over a range of hoop stresses (0-150 MPa)
Is there OPEX available for the fuel cladding between 500-600ºC?	No OPEX as fuel cladding
Is there OPEX available for the ducts between 350-500ºC?	No OPEX as ducts
Is the material ductile below 200ºC?	Limited knowledge – Tensile strength and DBTT show embrittlement at these temperatures from low dose (10-23 dpa). It should be noted that the magnitude of embrittlement is less than HT9 FM steel.
Is the supply chain available?	Yes

8 Oxide Dispersion Strengthened (ODS) Steels

99. There have been two main approaches to the development of improving a steel's performance (increase the high temperature limit, minimise low-temperature embrittlement, reduce swelling and increase tensile strength without sacrificing ductility) in a neutron radiation environment. These approaches are:

- Optimise the thermo-mechanical process (i.e. the manufacturing route).
- Attempt to engineer high sink strength properties by introducing high density of recombination centres through the creation of nanometre scale oxide dispersion throughout the material [43]. A sink is defined as a region where radiation damage can recombine, thus recovering the material structure.

100. For long-term operation (>15-20 years), the main objectives are to construct a steel that can withstand very high doses (200+ dpa) at high temperature (600-800ºC) and produce insignificant degree of swelling. The most promising approach is based on using powdered metallurgy to produce high strength, radiation resistant ODS steels [19, 44] (also named nanostructured steels within the literature).

101. China [43, 11], France [43], India [45], Korea [11], Russia [25], USA [19] and Japan [39] have all expressed interest in the development of ODS materials for SFRs to improve BU and increase operation temperatures.

8.1 Fundamental Properties of ODS Steel

102. The first ODS commercial steel to be developed was called MA957 by INCO [46] and is defined by the US patent 4,075,010. The alloy was developed to be the fuel cladding material for fast breeder reactors. The composition of MA957 is based on Fe-14Cr-0.25Y_2O_3 and is listed in Table 7.

103. ODS steels that contain 12-16 wt% Cr present high tensile and creep strengths over a large temperature range (500-800ºC) [19], unprecedented long-term thermal stability up to 900ºC [47] and high irradiation tolerance [48].

104. These favourable properties generally derive from the inclusion of small volume fraction (between 0.25 and 0.5 % Y_2O_3) with high density (5×10^{23} nano-oxides/m_3) and an average diameter of 2 - 8 nm that is dispersed uniformly across the material [44].

105. These nano-scale particles are generally based on yttrium-titanium-oxygen ($Y_2Ti_2O_7$) pyrochlores but have known to form Y_2TiO_3 and $YTiO_3$ [15].

106. These particles are highly stable under extreme operating conditions and act as recombination points for the vacancies and interstitials generated by radiation. This key property is the reason why ODS are named 'radiation resistant' in the literature.

107. The high temperature operating limit is generally determined by the amount of wt% Cr - the higher percentage gives higher creep resistance (up to 22 wt% before the delta brittle phase starts to form) [43].

108. Titanium (0.25 wt%) was added to later alloys to refine the nano particle size. [44].

109. Table 7 below shows two types of ODS steels currently investigated in France: 9 wt% Cr martensitic ODS steel and 14-18 wt% ferritic ODS steel [43].

110. The former 9 wt% has less corrosion resistance however manufacturing produces isotropic mechanical properties. The latter has more anisotropic properties and is susceptible to radiation-induced embrittlement. However, 14 wt% Cr steels have improved corrosion resistance and creep properties.

Table 7: Compositions of the most common ODS steels that are being investigated.

		Composition (wt %)										
Steel	**Country**	C	Cr	Ni	Mo	W	V	Ti	Si	Mn	N	Y_2O_3
MA957 [46]	USA	-	14	-	0.3	-	-	0.9	-	-	-	0.25
14YWT [49]	USA	-	14	-	-	3	-	0.4	-	-	-	0.3
12YWT [50]	USA	-	12	-	-	2.5	-	0.4	-	-	-	0.25
Fe-9Cr ODS [43]	France	0.1	9	0.15	-	1	-	0.3	0.3	0.3	-	0.3
Fe-14Cr ODS [43]	France	-	14	0.15	-	1	-	0.3	0.3	0.3	-	0.3
Fe-18Cr ODS [43]	France	-	18	0.15	-	1	-	0.4	0.3	0.3	-	0.5
EP450-ODS [51]	Russia	0.12	13.1	0.1	2.1	-	0.2	-	0.2	0.5	-	0.9

8.2 Manufacturing of ODS Steel

111. Due to the immiscibility of yttrium (Y) in iron (<0.002 wt% Y soluble in Fe), it is not possible to produce Y-containing steels via the traditional manufacturing routes (forging, casting and shaping). To introduce yttria (Y_2O_3) into the iron matrix, the alloy powder is mechanically milled before consolidation.

112. The general manufacturing process is as follows:

- Produce the base alloy material in the form of a fine-based powder.
- Mechanically mill yttria (Y2O3) powder with the fine-based alloy powder for 4 to 8 hours. This milling is carried out in inert atmosphere to reduce oxidation.
- Pour the powder into a can and degas in vacuum.
- Hot-isostatic pressing of the can between 900–1300ºC and 100-200 MPa for about 4 hours or extrude to produce the tubing form (i.e. cladding tubes).
- Post heat treatment is conducted to optimise the mechanical properties.

113. ODS manufacturing techniques have not been standardised with many research groups around the world investigating their own methods. The available data on the performance of ODS steel is fragmented as they vary in composition, manufacturing parameters leading to microstructural differences and batch-to-batch properties [52].

114. ODS steels have been made into tubes with the following drawbacks:

- Batches typically have low fracture toughness at room temperature.
- Fabrication of components with isotropic mechanical properties has not been demonstrated.
- The quality of experimental heats (production batches) is highly variable.
- The size of heats is in the order of kilograms.
- No demonstration yet of developing this manufacturing process to an industrial scale [52].

8.3 Welding ODS Steel

115. Welding is outside the scope of this research project; however, it should be noted as it is a significant challenge for ODS steels.

116. ODS nanoparticle's dispersion has been known to change under traditional fusion welding techniques, such as tungsten insert gas or electron beam welding. Using these techniques have shown to reduce the strength of the steel up to 70-80% [53].

117. Friction stir welding has shown better results for joining ODS to a ferritic alloy. The technique reduced the ODS's strength by 50%. This welding technique is still under heavy research [54].

118. It has been shown that under optimal conditions the nanoparticle distribution remains unaffected by the friction stir weld between ODS and F82H alloy (a steel similar to T91 FM steel) [54].

119. There are still significant issues with ODS welding: introduction of pores, changes in mechanical properties, etc. Long term performance of welds has yet to be demonstrated.

8.4 Operational Experience

120. Majority of the ODS steel's irradiation has occurred as test specimens and no study has investigated the synergistic effects (coolant compatibility, corrosion, irradiation damage, creep etc.) [12].

8.4.1 Tensile Stress

121. McClintock et al. [49] studied the mechanical properties of both 14WT (non-ODS steel) and 14YWT (ODS steel) in both unirradiated and neutron-irradiated conditions. These compositions are presented in Table 7.

122. Samples were irradiated in the High-Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory to a nominal fluence of 2.1×10^{21} n/cm_2 (1.5 dpa) at 300ºC, 580ºC and 670ºC. The tensile specimens were manufactured in the L orientation as specified in ASTM E399-06. They were miniaturised tensile specimens due to limited cost, time, space and activation in HFIR.

123. The tensile stress is shown in Figure 14. It is clear from the study's results that unirradiated 14YWT tensile stress (1435 MPa) is significantly higher than its unirradiated non-ODS variant (743 MPa) at room temperature. The study demonstrated that there was no appreciable hardening observed for 14YWT at 300ºC irradiated to 1.5 dpa. In addition, irradiated to 1.5 dpa at 670ºC, 14YWT demonstrated good resistance to irradiation-induced embrittlement.

124. The authors have meticulously detailed the specimen geometry, kept to international standards and correctly documented the testing environment. The literature reporting on ODS steel is typically incomplete hence the paper by McClintock et al. [49] is an example of a study that is well documented.

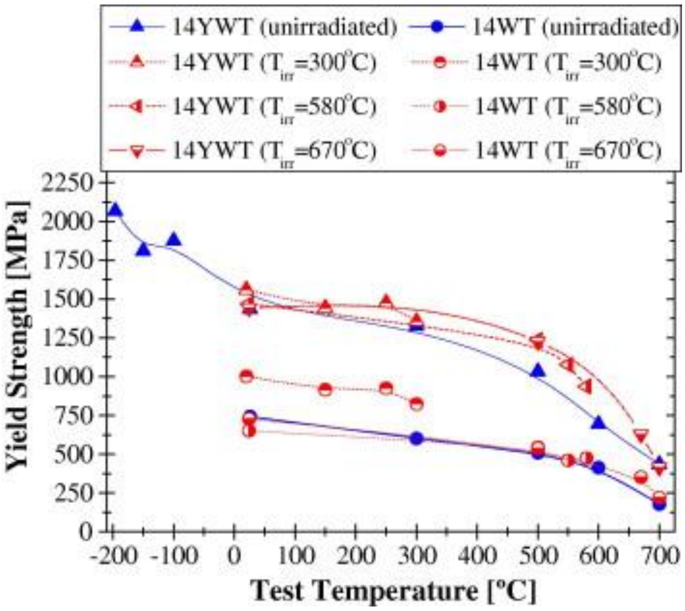

Figure 14 Tensile (yield) stress of 14WT and 14YWT irradiated to 1.5 dpa with the HFIR reactor at 300, 580 and 670ºC [49]. Notice the little change in stress in 14YWT steel (triangles data points) when irradiated; this is where the term 'radiation resistance' as the stress increases with an insignificant amount.

8.4.2 Impact Data (DBTT)

125. ODS steels are known to have an inferior DBTT compared to HT9 and T91 steels [19]. Tubes of ODS material manufactured by extrusion generally have considerably elongated grains along the extrusion direction that contribute to the increase in DBTT.

126. It is difficult to assess the impact data for ODS steel. The variability in manufacturing ODS steels impacts the mechanical properties and produces scattered DBTT data.

8.4.3 Fracture Toughness

127. The fracture toughness of ODS steel has been poor compared to HT9 and T91 [55] in the unirradiated conditions, as shown in Figure 15. The main cause of the lower fracture toughness is the unique ODS microstructure that produces shallow plasticity layers at high temperature allowing for cracks to propagate.

128. It should be noted that there is no consensus on the fracture toughness properties of ODS steel because the composition and manufacturing routes have not been finalised [19]. This produces difficulties in assessing against the criteria set out in this project.

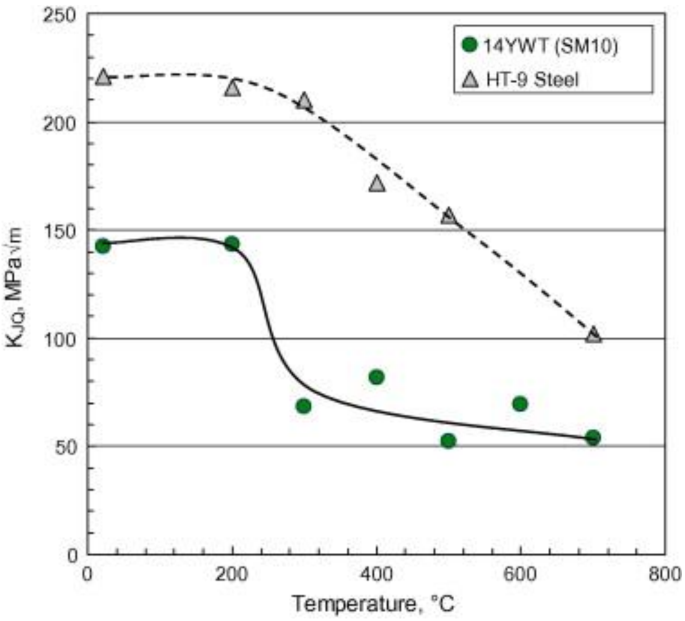

Figure 15 A direct comparison of the elastic linear fracture toughness as a function of temperature between unirradiated 14YWT ODS and unirradiated HT9 steels. Figure reproduced from [55].

8.4.4 Creep and Swelling

129. There have been no significant creep lifetime studies of ODS Steel in a radiation environment. Zinkle et al. [19] have complied the unirradiated thermal creep lifetime at 650ºC for ODS and FM (HT9 and T91 included. ODS steel's thermal creep resistance is greater than FM steels. This suggests that ODS steels can operate at a higher temperature under a specified operating stress.

130. There have been no long irradiation studies on swelling in ODS steels [19]. To provide an indication of the swelling resistance, multiple high dpa studies (100-500 dpa) have been conducted with single ion irradiation and have shown 1-3% volumetric swelling at 400 dpa. It should be noted that ion irradiation produces different damage to neutron irradiation hence the results should be considered carefully.

8.5 Future Use of ODS Steel

131. When considering the long-term future deployment of SFRs, it is clear from the literature that ODS steels are regarded as the likely (>15-20 years) fuel cladding material that could permit increases in the BU and operating temperature.

132. ASTRID, the French SFR that is in construction today, has decided to develop Fe-9Cr, Fe-14Cr and Fe-18Cr based ODS alloys to achieve up to 150-200 dpa without failure. The development of these materials is by CEA in collaboration with EDF and AREVA (now FRAMATOME) companies [43]. Further, CEA has a bilateral collaboration with Japanese Atomic Energy Agency (JAEA) to produce such ODS steel that is suitable for industrial production [18].

133. India has initiated a R&D program to investigate the use of ODS as cladding material for the future 500 MWe Prototype Fast Breeder Reactor (PFBR) [45]. The motivation for use of ODS steel is to allow for the higher BU.

134. Japan's research and development of ODS steels started in late 1990s to produce fuel cladding tubes that can withstand up to 250 dpa and temperatures as high as 700ºC. JSFR [39] is a commercially viable SFR currently in design and has selected both Fe-

9Cr and Fe-12Cr ODS steels as the primary candidate cladding materials. JAEA have irradiated 18 fuel pins with ODS cladding in the Russian BOR-60 [56].

135. The major challenges faced by JAEA were at low doses (51 dpa) where one fuel pin ruptured [56]. The premature rupture was due to the heterogeneous distribution of nano-oxide particles that produces heterogeneous mechanical properties. This result indicates that the ODS steels manufacturing process requires improvements to produce reliable mechanical properties.

136. The Russians have established a proven line of SFRs through their BN-series. To achieve economic viability (compared to the WWER reactor) they are considering the potential for changing the fuel cladding to ODS steel to increase BU [25].

137. Since 2005, the Russians have used the base alloy EP-450 and manufactured EP-450-ODS version. EP-450-ODS samples are currently being irradiated in BN-600 to an expected dose of 140 dpa between 375-700ºC since 2010 [57].

138. USA started the development of ODS steels for their fast breeder program that focused on MA957 alloy. Two ODS steels have now been identified as possible candidates for fission and fusion which are 14YWT and 12YWT [19]. The composition of these steels is listed in Table 7.

139. Within the UK, majority of ODS development is conducted in the Department of Materials at the University of Oxford. The university researches the manufacturing, welding, mechanical properties, neutron irradiation effects and collaborates with USA (Oak Ridge National Laboratory, Idaho National Laboratory), UK Atomic Energy Authority, and the International Thermonuclear Experimental Reactor (ITER).

140. In summary, there is significant research and development of ODS steel around the world. It is important to note that most of the development is conducted in national laboratories or vendor companies (for example EDF in France).

141. The real potential of using this material in actual reactors will become more apparent over the next two decades (>15-20 years).

8.6 Identification of Knowledge Gaps

142. ODS steels are one of the first approaches to design a steel that can resist radiation damage, have higher creep resistance and operate at a higher temperature. It is clear that there is significant amount of R&D still required to enable this steel to be classified as an engineering material (i.e. to be considered useable in a working environment). The identified knowledge gaps are:

- Manufacturing produces batch-to-batch variability in mechanical properties. For example, the mechanical properties could change through an axial position of a cladding tube. This is an unacceptable feature that yields the material unusable in an engineering capacity.
- No industrial manufacturing unit currently exists.
- Highly experimental steel that has no OPEX.
- No knowledge on how high radiation damage (50 dpa +) effects swelling, fracture toughness, impact properties and tensile stress.
- Welding ODS steel to non-ODS steel is still a significant challenge.
- ODS compositions have a large range indicating that the research and development has not concluded on a final composition.

143. Nevertheless, there is significant amount of effort going on for filling each of the above knowledge gaps. It is therefore reasonable to expect that ODS could be introduced in reactor designs over the next few decades.

8.7 Assessment of ODS Steel

144. The criteria outlined in Section 2 are used to assess the suitability of using ODS steels in a large scale SFR reactor. The outcomes are listed in Table 8.

145. The findings suggest the material is currently unsuitable for a reactor core/fuel. Nevertheless, the trends in literature are clear that this steel is under intensive development. The fundamental properties (high tensile stress, high temperature creep resistance and swelling resistance) suggests ODS steels have sufficient potential to become the fuel cladding and ducts of future (15-20> years) reactor designs.

Table 8: Assessment for the suitability of using ODS steel as fuel cladding and ducts for the SFR reactor core against the criteria outlined in Section 2.

Criteria	Assessment
Can the material survive >100 dpa?	No OPEX available.
Is the swelling tolerable?	No OPEX available.
Is there OPEX available for the fuel cladding between 500-600ºC?	No OPEX available. Test EP-450-ODS cladding has been inserted (2010) in the Russian BN-600 and is expecting to receive a dose of 140 dpa between 375-700ºC.
Is there OPEX available for the ducts between 350-500ºC?	No OPEX available.
Is the material ductile below 200ºC?	Limited – Test specimens irradiated at low dose (1.5 dpa) show minimal low temperature embrittlement.
Is the supply chain available?	No

9 Selection of materials for proposed SFR reactors

146. The developers of new reactor designs have released limited public information on the material selection and environmental conditions expected for the SFRs designed for deployment during the next 10 – 15 years.

147. HT9 steel is clearly the preferred material for SFR ducts and fuel cladding. The main reason for this is the significant amount of OPEX from EBR-II and FFTF [12].

148. The operational window of HT9 compiled from numerous studies is presented in Section 6.1, Figure 3. This window provides an opportunity to identify any major knowledge gaps in the publicly released SFR design plans.

149. Figure 16 clearly displays the major knowledge gaps that have been identified for using HT9 steel based on the publicly available SFR designs. A common trend in the publicly available information is that to increase the economics of an SFR, the time fuel is irradiated in the reactor must increase [43, 11, 45]. Consequently, the ducts and fuel cladding material will potentially receive doses that are beyond 147 dpa (the current knowledge limit on mechanical properties; swelling is known up to 208 dpa at 400ºC).

150. The knowledge gaps outlined in Figure 16 could provide a broad overview of the operational space that is understood and on the area which is not understood well. It should be noted that the values of dpa for SFR designs are the total fluence that is expected the material will be exposed to throughout the lifetime of the component.

151. It should be noted that T91 and ODS steels do not have sufficiently populated operational windows to provide a useful, sound investigation that would allow to identify the knowledge gaps in a similar way similar to that of Figure 16.

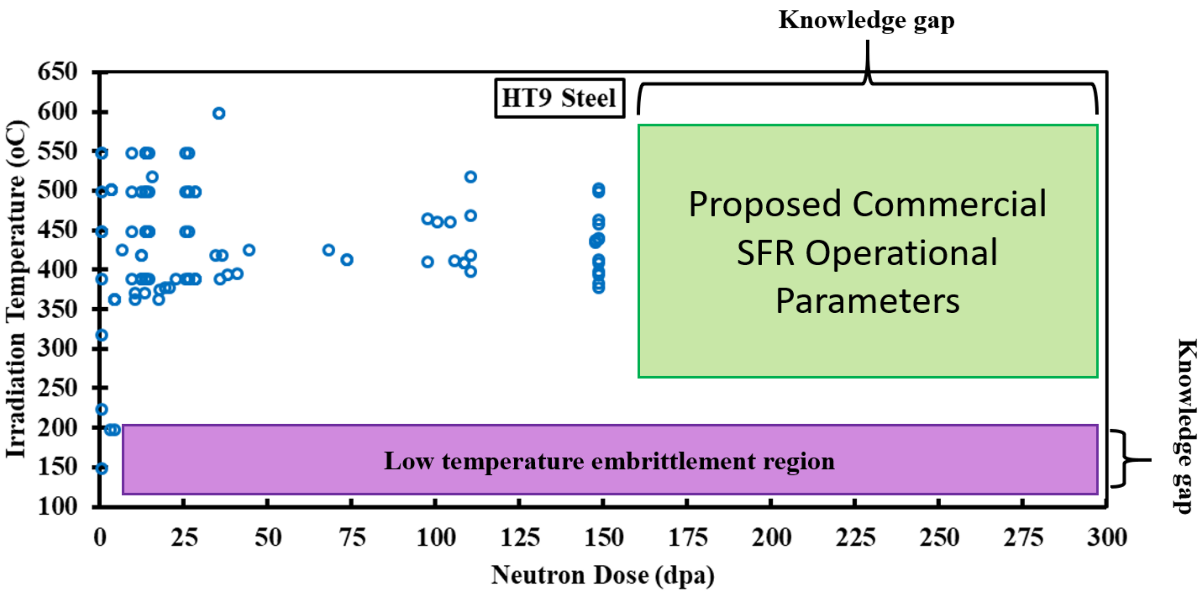

Figure 16: Operational window of HT9 steel (from Figure 3) with proposed SFR reactor designs that are in the public domain. It is clear where the major knowledge gaps are: beyond 160 dpa and low temperature regions (<200ºC).

10 Fuel Assembly Components

152. HT9 FM, T91 FM and ODS steels could be used in the fuel assembly's spacer grids and fuel rod's hold-down springs in current Pressure Water Reactors (PWRs).

10.1 Fuel Assembly Spacer Grids

153. The primary functions of the spacer grids are to hold the position of fuel rods, enhance critical heat flux and maintain the appropriate rod-to-rod distance. The spacer grids are located at different positions along the fuel assembly.

154. Functional requirements for the spacer grids in PWRs are:

- Space the fuel rods laterally and axially.
- Limit the fuel rod movements during operation and fuel handling operations.
- Minimise assembly bowing.
- Minimise the fuel rob vibration.
- For the bottom grid locations, act as a debris catcher.
- Resist the crushing lateral seismic forces in PWRs.

155. The mechanical design requirements are [60]:

- Provide adequate crush strength under accident conditions.
- Design should be compatible with shipping and handling loads.
- Corrosion and hydriding of materials should be understood, quantified and satisfy qualified irradiation tests.
- Springs within the spacers' materials performance should satisfy the elastic properties and creep strength.

10.2 Fuel Rod Plenum Springs

The fuel rod inside the assembly is a tube made of in zirconium alloy (PWR) or ferritic/martensitic steel (SFRs) which contains a stack of fuel pellets. Some space (plenum) is left on the top of the rod to accommodate fission gases. The plenum spring holds down the fuel during shipping and handling.

156. The functional requirement for the plenum spring within the fuel rod is to eliminate pellet stack movement during handling, transport and fuelling the reactor- so that pellet chipping and gaps between pellets are sufficiently reduced or eliminated.

157. The mechanical design requirements for the hold-down are:

- Must maintain proper positioning under normal operating conditions and in Design Basis Accidents (DBAs).
- Provide a plenum space for the released fission gases.
- Compress sufficiently to accommodate the fuel pellet expansion.
- Provide satisfactory stress relaxation properties.

158. The essential functional requirements and mechanical design requirements to the core components depend on the overall reactor design which sets the operational environment for fuel assemblies, e.g. for the water cooled PWR: pressure of 155-158 bar, average temperature of 280 - 300ºC, coolant flow of 3-6 m/s, average power rating 80-125 kW/l and average fast neutron flux of 6-9×1013 ncm-2s-1 [58]. For comparison, the test SFR BOR-60 assemblies operate with: sodium inlet temperature 310-340ºC and outlet temperature 530ºC, flow velocity of 8m/s and neutron flux density 3.6×1015 ncm-2s-1 [59].

159. PWR irradiation conditions are not as challenging for swelling as the SFR conditions and do not mandate a material change for current fuel irradiations. However, it is considered that regardless of the differences in environment, using new materials in PWR operations could provide a cost-effective way for gathering useful irradiation data on the materials that could be used in SFRs.

11 Conclusions

160. The objective of this report is to identify the major gaps in the publicly available knowledge regarding the properties, applicability and availability of iron-based materials for advanced nuclear technology – using SFRs OPEX as example.

161. A set of specific criteria have been established - based on the review of various commercial and test SFRs and aiming to provide broad assessment of the materials that could be selected for future reactor. The following material properties are considered fundamental: radiation damage limit, swelling, operational temperature limit, low-temperature embrittlement threshold and manufacturing capability.

162. The criteria are:

1) Can the material survive beyond 100 dpa?
2) Is the radiation-induced swelling tolerable?
3) Is there OPEX available for the fuel cladding between 500-600ºC?
4) Is there OPEX available for the ducts between 350-500ºC?
5) Is the material ductile below 200ºC?
6) Is the supply chain available to produce the material?

163. The main findings of the candidate materials selection against the criteria are:

164. Austenitic Stainless Steels:

- For test reactor operational envelope, austenitic stainless steels have sufficient OPEX and minimal knowledge gaps.
- For commercial reactor operational envelope, austenitic stainless steels exhibit a significant radiation-induced swelling challenge.
- The swelling is inherent to the material and either the material needs to change or the operational envelope needs to be brought closer to that of a test reactor.
- The knowledge gaps are narrowed below 100 dpa (due to the swelling) as there if a wealth of OPEX. However, beyond 100 dpa the knowledge gaps are widened as this operational area is rarely explored.

165. HT9 Ferritic/Martensitic Steel:

- This is the leading candidate for the fuel cladding and duct structural material in a commercial SFR – as demonstrated by a number of existing designs.
- Significant OPEX is available of fuel cladding and ducts within the temperature range applied in commercial SFRs.
- Radiation-induced swelling is tolerable.
- Knowledge gap of irradiated HT9's mechanical properties >100 dpa; <100dpa, HT9's tensile stress, shift in DBTT and fracture toughness is well understood. The knowledge on the ductility of HT9 below 200ºC is therefore limited.
- No supply chain worldwide; no ASTM standard available; not adopted by ASME BPVC code yet.
- HT9 steel's knowledge gap is minimised for test SFR operational envelope.
- For a commercial reactor envelope, HT9 steel's knowledge gap is widened. The size of this gap will depend on the operational envelope proposed.

166. The main findings of the new materials selection against the criteria are:

167. T91 Ferritic/Martensitic Steel:

- Near-term (>10 years) candidate as the fuel cladding and duct structural material in a commercial SFR.
- T91 has the potential to operate at higher temperatures (650-700ºC) and exhibits less low-temperature embrittlement compared to HT9.
- No OPEX as fuel cladding and ducts within the temperature range specified.
- Radiation-induced swelling tolerable.
- Knowledge gap of irradiated T91 steel's mechanical properties (>26 dpa for shift in DBTT, >23 dpa for tensile stress and >100 dpa for fracture toughness known). Therefore, limited knowledge on the ductility of T91 below 200ºC.
- Supply chain is available worldwide; established in ASTM standards; adopted by ASME BPVC code.
- For either test or commercial reactor envelopes, T91 steel's knowledge gap is significant and needs to be addressed.

168. ODS Steels:

- Long-term (>15-20 years) candidate as the fuel cladding and duct structural material in a commercial SFR.
- ODS steels potentially can operate between 600-800ºC, withstand (i.e. minimal embrittlement) very high doses (200+ dpa) and exhibit minimal swelling.
- No OPEX as fuel cladding and ducts within the temperature range specified.
- Radiation-induced swelling is tolerable.
- No industrial manufacturing unit exists worldwide.
- Manufacturing produces batch-to-batch variability in mechanical properties and is currently unsuitable for an engineering material.
- Test specimens inserted into test reactors have produced the currently available dataset on ODS's performance in irradiating environments.

169. The likelihood of bridging the knowledge gaps will depend on increasing the OPEX. Specifically, irradiating ducts and fuel cladding in test reactors to build OPEX up to similar levels as HT9 FM will be required. One method to accumulate additional OPEX is to use new materials for the PWR spacer grids and plenum springs.

170. Another lesson to learn from FFTF is that any new material considered must have previous OPEX to provide a board understanding of the behaviour of this material in the required operational environment.

171. In the author's opinion regarding new materials for the cores 'Generation IV' reactors, a significant amount of testing is required to determine whether the material is suitable for the designed application. The testing should consist of inserting the new material as components and test samples in test or operating reactors under the correct operational conditions to capture synergistic effects. The role of test nuclear reactors is essential to providing sound scientific data that will influence future reactor designs.

172. It should be noted that the opinions of the author expressed in this report do not represent ONR or their regulatory opinion.

12 Appendix

12.1 Definitions of Material Science Terms

174. Within this report many material science terms are used. In this section, a brief definition is provided to bring clarity to the reader.

Tensile Strength

175. Tensile strength is the capacity of a material to withstand loads that tend to elongate when loaded in simple tension. This can be expressed in terms of:
- the yield stress - the stress at which appreciable plastic deformation first occurs is known as yield stress.
or
- the ultimate tensile stress - the value of tensile stress at which the material deformation is not likely to be arrested by strain hardening before failing occurs.

Ductile-to-Brittle Transition Temperature (DBTT)

176. Impact data consists of the total energy absorbed by a material for it to fracture (over a range of temperatures). This absorbed energy is dependent on the failure mechanism.

177. bcc materials exhibit a shift in failure modes from brittle (low temperature) to ductile (high temperature) at a certain transition temperature, named DBTT. This phenomenon does not occur in fcc materials due to the crystal structure's many slip planes which are active over all temperature ranges (allowing for only ductile failure). Whereas bcc's slip plans are inactive at low temperatures (allowing for only brittle failure) and are active at high temperatures (allowing for ductile failure).

178. Often, to determine this DBTT, a notched specimen is used in a Charpy Test. A hammer impacts the notched specimen with a known energy and determines how much energy is used to break the specimen. A higher energy absorbed indicates a ductile failure and the lower energy absorbed indicates a brittle failure. Examining the fracture surface provides verification.

179. Neutron radiation is known to reduce the energy required to produce ductile failure due to embrittlement (called the upper shelf-energy) and DBTT increases. The DBTT before and after irradiation for HT9 ferritic/martensitic steel (bcc crystal structure) is shown in Figure 17. The shift in DBTT is normally quoted to indicate the effect of irradiation damage.

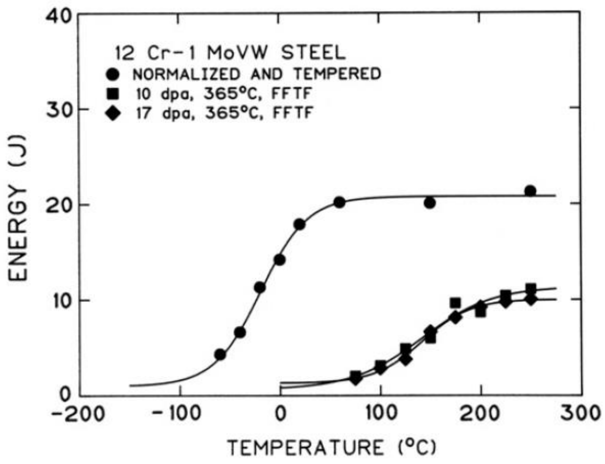

Figure 17: Charpy impact curves (energy absorbed to cause failure) for HT9 FM steel in both the unirradiated and irradiated conditions over a range of temperatures. It is clear that irradiation reduces the upper self-energy and shifts the DBTT towards higher values [12].

Swelling

180. Radiation damage produces vacancies which cluster to form voids (vacuum spaces) within the material. (n, alpha) reactions can occur in nickel additions in the steel and, less common, iron. The alpha particles stabilise the voids and form helium bubbles with an associated internal pressure. This pressure expands the lattice producing a volume change. For further information on the swelling, see Appendix 12.2.

Creep

181. Creep is diffusion-controlled plastic deformation (shape change) which occurs over time when a stress is applied. Under elevated temperatures diffusion rates are increased therefore creep rates are sensitive to temperature. Creep strain is generally associated with damage to the materials crystal structure which enables materials to fail at loads below their yield stress.

182. Radiation damage introduces vacancies within the lattice which aid the diffusion of dislocations (2D lattice defects) around barriers, leading to more dislocations slipping. Radiation damage can induce and/or enhance creep in alloys.

Fracture Toughness

183. The value of fracture toughness describes the ability of a material to resist fracture by crack propagation. The fracture toughness of a material is related to the stain energy released when a crack grows. This is expressed in terms of the critical stress intensity. There are two types of fracture toughness values, depending on whether crack growth is assumed to involve significant plastic deformation: the linear-elastic fracture toughness factor, K, and the elastic-plastic fracture toughness, J. The latter is determined by solving the J-integral from several points along a loading curve that includes plastic deformation. K is used for brittle failure and J-integral for ductile failure.

Displacements per atom (dpa)

184. The definition of dpa is the number of times on average an atom has been displaced during a given period of exposure to radiation. When a material is exposed to 1 dpa then on average each atom has been displaced at least once. Radiation damage models are used to convert from instantaneous neutron fluence and neutron energy spectrum to dpa and are outlined in ASTM E693-94 [61].

185. The unit dpa provides an indication of how much a material has been exposed to radiation damage.

Burn Up (BU)

186. The total amount of energy produced per unit fuel (BU) is proportional to the fuel material depletion due to irradiation. Fuel material burnup and structural material exposure (measured in dpa) are closely related.

12.2 The Nature of Swelling in Materials

187. An incident neutron hitting an iron lattice will displace atoms producing an interstitial (the displaced iron atom must stop somewhere) and vacancy (vacant lattice site). For austenitic stainless steels and in the temperature range of 300 – 700 ºC these vacancies start to cluster and form voids.

188. Helium produced by the (n,α) reaction with mainly nickel (Ni) stabilises these voids. Under constant irradiation and temperature, these voids start to grow. The nuclear reaction is:

$$^{58}\mathrm{Ni}(n,\gamma) \rightarrow {}^{59}\mathrm{Ni}(n,\alpha) \rightarrow {}^{56}\mathrm{Fe}$$

where α is a helium nucleus.

189. Void formation and growth are sensitive to chemical composition, manufacturing defects and microstructure, dose, dose-rate, irradiation temperature, stress state and

the history of time-varying temperatures. There are two stages to swelling [2]: transient regime and steady-state regime.

190. The transient regime has been found to be sensitive to the following factors:

- Initial dislocation network density. For a solute-free alloy the dislocation network's 'sessile' (immovable) loop-dominated microstructure un-faults to a glissile (movable) dislocation state. This transition in the dislocation network is when the transient transforms to a steady state swelling rate regime [11].
- Cold work introduces extra dislocations to the microstructure increasing the density to a value which is too high for void nucleation. The dislocation network must relax before the swelling regime can initiate. This relation will be met once the alloy has received certain significant amount of irradiation dose.
- In engineering alloys, the high diffusive elements (silicon (Si) and phosphorus (P)) suppress void formation. This is facilitated by the fact high diffusive elements increase the diffusivity of vacancies that reduces the super-saturation of vacancies which drives void nucleation.
- Under irradiation, P, Si and Ni undergo irradiation-induced segregation. This precludes them from contributing to vacancy diffusion, thus eliminating their impact on the void nucleation mechanism.
- Additions of Ti and Nb produce TiC and NbC within the steel microstructure, respectively. These particles act as re-combiners of point defects by allowing vacancies and interstitials to meet. The reduction of vacancies reduces the onset of void nucleation.

191. For FM steels, the ferritic and martensitic microstructures have high dislocation density which improves greatly the transient region of swelling [12]. Additionally, the vast reduction of Ni in these steels reduces the generation rate of helium that stabilises the voids. These two main processes provide FM steels with great swelling resistance.

12.3 Complete Table of Findings

Table 9 The major findings of the material selection for current and new materials for ducts and fuel cladding in SFR reactors against the criteria established in this report.

	Key findings of:		
Material	**Operational Experience**	**Material's Limitations**	**Industrial availability**
Austenitic Stainless Steels (316, 316Ti, 15-15Ti, D9)	- Nearly all SFR cores were constructed out of austenitic stainless steels. - Wealth of OPEX from DFR, PFR, EBR-I, EBR-II, FFTF, Phénix, Superphénix, BOR60, BN-series, JOYO, Monju, FBTR, PFBR.	- Intolerable swelling of all austenitic stainless steels which render them unusable as fuel cladding (50-100 dpa). - Modifications, such as adding Ti, delay the onset of swelling. They ultimately swell given enough radiation dose (50-100 dpa).	- Extensive manufacturing availability for all austenitic stainless steels. - Used throughout nuclear power plants (steam generators, pressurisers, piping, heat exchangers).
HT9 Ferritic Steel	- Leading candidate for ducts and fuel cladding in SFRs. - EBR-II: Test cladding and test specimens from 1960s to 1994 - FFTF: Test cladding and fuel assembly from 1982 to 1992. - BOR-60: test specimens only.	- Swelling tolerable <208 dpa - Operated between 350-550ºC successfully. - Knowledge gap in low temperature embrittlement (<200ºC). - No mechanical data >160 dpa. - Operated 550ºC max; 200ºC min data available - Shift in DBTT known up to 147 dpa.	- No manufacture currently available worldwide. - Defined in ASTM A771. - No HT9 in ASME BPVC Code Section III Subsection NH – Class 1.
T91 Ferritic-Martensitic Steel (Mod 9Cr-1Mo)	- Near-term (>10 yrs) candidate. - EBR-II: test specimens only. - FFTF: test specimens only. - BOR-60: test specimens only. - Phénix: test specimens only.	- No OPEX as fuel cladding and ducts. - No data on mechanical properties beyond 26 dpa - Swelling tolerable <208 dpa. - Maximum operating temperature 550ºC - No long term (>10yr) creep data.	- Industrial scale manufacturing available. - Defined in ASTM A213 A182 and A335. - Produced today for coal power industry.

Oxide Dispersion Strengthened (ODS) Steel	- Long-term (>15-20 yrs) candidate. - BOR-60: test specimens only. - HIFR: test specimens only. - BN-600: experimental fuel cladding since 2010.	- No OPEX as fuel cladding and ducts. - Shown limited ‘radiation resistance’ properties. - Unsatisfactory amount of data available to provide confidence in limitations.	- No worldwide manufacture available - Batch-to-batch variability in mechanical properties - Heavy R&D on going. - Welding ODS steel to non-ODS steel is still a significant challenge.

12.4 Raw data for HT9 and T91 Steel's Mechanical Properties

Table 10 The complete dataset used in this report.
The source of the data is given for each value.

Material	Temperature (ºC)	Radiation Dose (dpa)	Shift in DBTT (J)	Fracture Toughness (kJ/m2)	Tensile Stress (MPa)	Source
HT9	200	2.47	169			[10]
HT9	200	3.7	163			[10]
HT9	365	4	129			[10]
HT9	365	4	130			[10]
HT9	365	4	134			[10]
HT9	365	4	157			[10]
HT9	427	6	108			[10]
HT9	365	10	160			[10]
HT9	390	12	127			[10]
HT9	390	13	124			[10]
HT9	390	13	89			[10]
HT9	450	13	27			[10]
HT9	500	13	33			[10]
HT9	550	13	57			[10]
HT9	550	13	9			[10]
HT9	365	17	160			[10]
HT9	390	26	144			[10]
HT9	450	26	59			[10]
HT9	500	26	43			[10]
HT9	550	26	41			[10]
HT9	420	34	87			[10]
HT9	420	36	107			[10]
HT9	400	110	157			[10]
HT9	420	110	97			[10]
HT9	470	110	52			[10]
HT9	520	110	85			[10]
HT9	380	148	210			[30]
HT9	395	148	205			[30]
HT9	400	148	215			[30]
HT9	385	148	220			[30]
HT9	410	148	170			[30]
HT9	415	148	180			[30]
HT9	440	148	86			[30]
HT9	442	148	88			[30]
HT9	465	148	80			[30]
HT9	460	148	95			[30]
HT9	500	148	70			[30]

Material	Temperature (ºC)	Radiation Dose (dpa)	Shift in DBTT (J)	Fracture Toughness (kJ/m2)	Tensile Stress (MPa)	Source
HT9	505	148	60			[30]
HT9	390	0			540	[32]
HT9	450	0			500	[32]
HT9	500	0			500	[32]
HT9	550	0			420	[32]
HT9	390	9			675	[32]
HT9	450	9			510	[32]
HT9	500	9			515	[32]
HT9	550	9			430	[32]
HT9	373	9.8			810	[27]
HT9	373	12.8			820	[27]
HT9	390	13			886	[31]
HT9	390	13			880	[31]
HT9	450	13			610	[31]
HT9	450	13			610	[31]
HT9	500	13			550	[31]
HT9	500	13			570	[31]
HT9	550	13			480	[31]
HT9	550	13			480	[31]
HT9	390	22.2			785	[27]
HT9	390	25			880	[31]
HT9	450	25			570	[31]
HT9	500	25			525	[31]
HT9	550	25			480	[31]
HT9	390	35.3			790	[27]
HT9	427	44			610	[27]
HT9	427	67.5			620	[27]
HT9	25	0		97		[34]
HT9	150	0		95		[34]
HT9	225	0		45		[34]
HT9	320	0		50		[34]
HT9	450	0		70		[34]
HT9	550	0		110		[34]
HT9	55	5		63		[34]
HT9	55	5		52		[34]
HT9	55	5		57		[34]
HT9	390	12		96		[34]
HT9	390	12		76		[34]
HT9	420	12		95		[34]
HT9	420	12		62		[34]
HT9	500	12		55		[34]
HT9	390	14		67		[34]

Material	Temperature (ºC)	Radiation Dose (dpa)	Shift in DBTT (J)	Fracture Toughness (kJ/m2)	Tensile Stress (MPa)	Source
HT9	450	14		61		[34]
HT9	450	14		80		[34]
HT9	500	14		82		[34]
HT9	550	14		87		[34]
HT9	520	15		107		[34]
HT9	390	26		53		[34]
HT9	390	26		58		[34]
HT9	500	26		64		[34]
HT9	390	28		57		[34]
HT9	390	28		69		[34]
HT9	390	28		65		[34]
HT9	500	28		72		[34]
HT9	500	28		72		[34]
HT9	600	35		75		[34]
HT9	600	35		75		[34]
HT9	415	73		81		[34]
HT9	415	73		40		[34]
HT9	411	108		90		[34]
HT9	411	108		53		[34]
HT9	503	3		112		[36]
HT9	503	3		125		[36]
HT9	503	3		106		[36]
HT9	503	3		79		[36]
HT9	467	97		79		[36]
HT9	463	100		101		[36]
HT9	462	104		135		[36]
HT9	439	147		94		[36]
HT9	438	147		85		[36]
HT9	437	147		89		[36]
HT9	414	105		67		[36]
HT9	412	97		96		[36]
HT9	397	40.3		58		[36]
HT9	395	37.6		61		[36]
HT9	380	20.1		68		[36]
HT9	379	18.8		54		[36]
HT9	377	17.4		51		[36]
T91	500	13	9			[40]
T91	500	26	11			[40]
T91	55	5.5	85			[40]
T91	55	10	135			[40]
T91	450	13	50			[40]

Material	Temperature (ºC)	Radiation Dose (dpa)	Shift in DBTT (J)	Fracture	Tensile Stress (MPa)	Source

				Toughness (kJ/m2)		
T91	450	13	70			[40]
T91	450	26	47			[40]
T91	390	13	55			[40]
T91	450	13	5			[40]
T91	500	13	8			[40]
T91	550	13	0			[40]
T91	390	26	55			[40]
T91	500	26	10			[40]
T91	200	2.5	109			[40]
T91	200	4	131			[40]
T91	22	0			547	[41]
T91	400	0			474	[41]
T91	500	0			438	[41]
T91	550	0			440	[41]
T91	390	10			781	[41]
T91	450	10			480	[41]
T91	500	10			445	[41]
T91	550	10			429	[41]
T91	390	10			890	[31]
T91	450	10			475	[31]
T91	500	10			450	[31]
T91	550	10			440	[31]
T91	390	23			740	[31]
T91	450	23			475	[31]
T91	500	23			430	[31]
T91	550	23			410	[31]
T91	55	5		33		[35]
T91	55	5		35		[35]
T91	55	5		31		[35]
T91	420	11		57		[35]
T91	420	11		43		[35]
T91	520	15		77		[35]
T91	520	15		49		[35]
T91	415	70		52		[35]
T91	415	70		47		[35]
T91	411	105		56		[35]
T91	411	105		71		[35]
T91	232	0		72		[35]
T91	427	0		78		[35]

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