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Abstract

Research is currently being conducted to find the optimal steel to use in the vacuum vessel (VV) of ARIES power plants. The VV should meet several design criteria, including fabrication and activation requirements. In this report, several different types of steels are examined. Their properties under irradiation and activation characteristics were the primary benchmarks used in determining the ideal material for the ARIES VV. Steels generating high-level waste were excluded from possible material choices from the beginning and all materials were to be recyclable (able to be reused within the nuclear industry). Based on these design criteria, it was found that the 3Cr-3WV reduced-activation steel is the best candidate for the ARIES VV as it is recyclable, qualifies as Class A low-level waste, and has the potential to satisfy the VV fabrication requirements.

1. Introduction

The primary functions of the vacuum vessel are to provide the high-level vacuum environment (necessary to achieve and maintain high-quality fusion plasma), support the internal power core elements, and, along with the blanket and shield, provide a shielding function to protect the magnets against radiation. In addition, the VV is a safety-class component that implements safety functions, such as confining the tritium and radioactivity and limiting the public/worker exposure to radiation during accidents. It is a robust, double-walled structure with ribs and bulk shielding materials between two 3-5 cm thick structural face sheets. Unlike the blanket and shield, the VV could operate at lower temperature (150-200^{o}C) to serve as a heat sink during LOCA/LOFA events. The preferred coolant for the VV is water to reduce fast neutron fluence to superconducting magnets [1,2] and help reduce the overall size of the power plant. Other activation-related requirements include [3]:

- Recyclable structural and filler materials
- Generate only low-level waste, preferably Class A to reduce disposal cost
- Generate low decay heat to act as a heat sink during LOCA/LOFA events.

F28H steel [4] as a VV material has the advantage of having a composition specifically tailored to facilitate long-term waste disposal and/or recycling after plant decommissioning. However, it was initially envisaged as a first wall/blanket structural material operating above the radiation hardening temperature regime as much as possible in order to avoid upward temperature shifts in the ductile-to-brittle-transition-temperature (DBTT). As a VV material, it would be operating at low temperature entirely in the radiation hardening regime and would be subject to shifts in the DBTT which could eventually result in operating the vessel in the lower fracture toughness regime. In addition, there is also the necessity for the post-weld heat treatments (PWHT) at ~750^{o}C during assembly of the VV.

Selection of an austenitic stainless steel would eliminate the DBTT issue and also eliminate the need for the welds to be tempered at 750^{o}C. However, there are several materials issues that would have to be addressed:

- Selecting a composition that would meet the activation requirements for maintenance dose, decay heat, and long term waste disposal (e.g. excluding Mo)
- Possible reductions in fracture toughness and loss of uniform strain as a result of damage accumulation > 10 dpa at ~200^{o}C
- Possible swelling issues at doses > 20 dpa in sections operating at temperatures > 350^{o}C.

These issues and others require careful evaluation from perspectives of VV fabrication, materials properties under irradiation, and activation. There is a wealth of information available to support a thorough evaluation of the various steel options.

2. VV Fabrication Issues

Previous ARIES designs employed the blanket/shield low-activation F82H ferritic steel for the VV without paying much attention to the choice of material. The adopted VV design approach for most ARIES designs was double-walled, low-activation ferritic steel (FS) filled with water and fillers (tungsten carbide or borated steel) as additional shielding materials. The VV structure should be weldable and reweldable after irradiation at any time during operation. Given the low operating temperatures for the ARIES VV, the Cr content of the F82H steel is satisfactory from the corrosion point of view. However, it would normally require heat treatment at 700-750^{o}C for 0.5 to 2 hours after welding to temper the martensitic structure and develop high toughness combined with a low DBTT. Because of its large size, it is unpractical for the VV to be heat-treated (tempered) following assembly and welding. Clearly, the necessity to temper the F82H FS at 700-750^{o}C presents difficult issues. For this reason, the F82H FS is unacceptable for the VV due to the complex heat treatment requirement.

Even though welding is a mature process, there are new innovations in welding, such as friction stir welding (FSW). Essentially, it is a solid-state joining process that does not radically alter the material composition and for some materials a post-weld heat treatment may not be required. It provides a high quality weld using quite low temperature for welding, thus the material properties do not degrade as much [5]. Even though FSW is currently limited to thin structures (< 3 cm thick), it may eventually be able to accommodate thick sections and should certainly result in less distortion and residual stress. However, initial results on F28H indicate that the weld region is transformed to austenite and subsequently transforms back to martensite on cooling and therefore a post weld tempering treatment is still required. In general, it is probably too early to assume that some form of stress relief treatment would not be necessary for any thick-section welded structure, austenitic or ferritic.

Besides the welding and PWHT issues, the expected neutron fluence and operating temperature ranges for the VV could place the F82H FS firmly in the radiation hardening regime. Irradiation of F82H at ~200^{o}C to a dose of ~1 dpa would result in an increase of the 200^{o}C yield stress of around 200-300 MPa and a corresponding upward shift in the DBTT. The magnitude of the shift in fracture toughness transition temperature would depend on extrinsic factors such as flaw geometry, section thickness, etc., but the evidence suggests the shift could be of the order of ~100^{o}C [6]. This could conceivably result in some parts of the VV operating in a brittle rather than a ductile fracture regime.

In summary, the steel structure of the ARIES VV has to meet the following fabrication requirements:

- Operable in and compatible with water cooling at low temperature (150-200^{o}C)
- Develop a safe, fracture-resistant microstructure with adequate strength and toughness, including all welded regions
- No substantial embrittlement at low operating temperature
- Easily reweldable with no need for complex Post-Weld Heat Treatment (PWHT)
- Tolerable neutron-induced swelling, particularly behind assembly gaps and near penetrations (> 20 dpa) [7].

3. Candidate Steels for VV Fabrication

For the ARIES VV, the shielding, activation, dose during maintenance, decay heat, and waste disposal, fabrication, and mechanical property requirements should all be carefully balanced. There is a class of low-activation steels that may not require extensive development and qualification programs. Some steels are currently used in the industry while others are still in the developmental stage. Table 1 provides a brief assessment of the known and unknown properties and fabrication aspects of the following six steels as well as F82H:

- 3Cr-3WV bainitic steels
- 8-9%Cr reduced activation ferritic-martensitic steels (ORNL-FS)
- 430 ferritic steel (430-FS)
- 316 SS austenitic steel (316-SS)

- Modified DIN-4970 austenitic stainless steel (RAAS)
- AMCR-0033 Mn-stabilized austenitic stainless steel (Mn-AS).

Of particular interest is the 3Cr-3WV steel. This is relatively new, reduced-activation steel originally developed at Oak Ridge National Laboratory (ORNL). In the 1990s, Klueh [8] investigated a series of 2-3 Cr ferritic steels based on the well-known 2 1/4 -1Mo steel widely used for water/steam piping applications. To ensure favorable long-term waste disposal characteristics, Mo, Ni, Nb and Cu were restricted to low levels and W and V were introduced for hardening/strengthening purposes. Several of these experimental alloys had good strength properties and for some alloys, the as-welded microstructure had very good toughness properties that would not require a post-weld heat treatment. The 3Cr-WV alloys subsequently attracted interest from the petroleum and chemical industries for reactor vessels and heat recovery systems. Compared to the widely used 2 1/4 Cr-Mo steels, these new alloys offer a combination of higher strength, reduced fabrication costs, a lower DBTT, a higher upper-shelf energy, ease of heat treating and a strong potential for not requiring a PWHT. Under a joint ORNL-industry partnership, several of these steels have been successfully scaled up to 50-ton heats and significant progress has been made towards full commercialization. An ASME Code Case has been granted based on the extensive database developed for one of the alloy grades.

Ferritic steels containing 14%-18% Cr (e.g., Type 430) remain ferritic throughout their operating range and therefore a PWHT is not required to temper a martensitic structure as for the 8-9% Cr steel. While the ferritic steels have good corrosion resistance in aqueous environments, it may not escape the necessity of having some measure of water chemistry control particularly in a radiation environment. There are a couple of issues with these steels that need more consideration. Firstly, welding thick sections can be a problem because of rapid growth of the ferrite grains that can lead to reduced fracture toughness and increased susceptibility to intergranular corrosion. Secondly, ferritic steels operating at 150-250^{o}C will be more susceptible to radiation hardening and shifts in DBTT than the 8-9% Cr reduced-activation FS.

It was to avoid these kinds of issues that ITER [9] selected austenitic steel 316LN-IG for its VV since there is no DBTT in such materials. However, austenitic steels are not free from radiation damage and activation issues. Neutron-induced swelling is a concern in VV sections behind assembly gaps and close to penetrations and ports where the atomic displacement could approach 20 dpa in ARIES designs. There is also the issue of radiation-induced loss of uniform strain and fracture toughness that occurs for 316-SS in a dose-temperature regime above ~5 dpa and temperatures of 250-350^{o}C. For lower temperatures the strain-to-necking values remain > 5% for doses at least up to 10 dpa. The low-temperature radiation effects in 316-SS have been mapped out and ITER has a good database. As discussed later, the 2.5wt% Mo of 316-SS causes activation problems and generates high-level waste. Rieth [10] proposed replacing Mo with W and made other modifications compared to 316-SS, aiming at producing a reduced-activation austenitic steel (RAAS). Even with a composition tailored to meet the activation requirement the use of RAAS does not solve the VV problems entirely. Depending on the operating condition, it may be necessary to assess the feasibility of radiation damage annealing treatment for RAAS to maintain uniform strain and fracture toughness at levels that would provide satisfactorily safe margins.

Table 1. Comparison of Fabrication Parameters of Candidate Steels

Name	MF82H	3Cr-3WV	ORNL 8-9% Cr	16-18% Cr	316-SS	DIN-4970	AMCR-0033
Type	**FS**	**Bainitic FS**	**RA F/MS**	**430-FS**	**AS**	**RAAS**	**AS**
Requires complex PWHT?	Y	N	Y	N	N	N	N
Corrosion resistant in 200^{o}C water?	Y	Y	Y	Y	Y	Y	Y
Need water chemistry control to inhibit IASCC?	TBD	TBD	TBD	TBD	Y	Y	Y
Radiation hardening and DBTT shift @ 150-200^{o}C, 10 dpa?	High	TBD	Smaller	TBD	#	#	#
Welding issues for 2 cm thick plates?	TBD	TBD	TBD	TBD	N	N	N
Thermal conductivity	High	High	High	High	Low	Low	Low
Swelling @ 10-20 dpa and 300^{o}C?	Low*	Low*	Low*	?	Low*	Low*	Low*
Thermal expansion	Low	Low	Low	Low	High	High	High
Relatively expensive?						Y	N

\# No DBTT, but reductions in uniform strain and fracture toughness.

* Less than 5%.

In the 1980s, fusion materials programs in the EU, US, and Japan investigated the feasibility of eliminating Ni and utilizing Mn as a means of stabilizing the austenitic structure in reduced activation stainless steels. For example, Piatti and Schiller [11] measured selected thermal and mechanical properties for four Cr-Mn (Ni-free) austenitic steels of interest for fusion applications and significant amount of irradiation data were developed in the US [12]. One obvious concern is the 17.5 wt% Mn in the AMCR-0033 steel. This could be problematic, as manganese tends to generate high decay heat compared to iron, resulting in higher VV temperatures during LOCA/LOFA [13].

4. Compositions of Candidate Steels

Alloying elements and impurities will definitely impact the activation level of steel-based components. For fusion, all materials are carefully chosen to minimize long-lived radioactive products such as C-14, Ni-59, Nb-94, Mo-99, Re-186m, etc. Manufacturing companies for fusion materials made a serious effort to use highly pure raw materials, strove to exclude Mo and Re, and minimize Nb impurities in particular. Table 2 lists the alloying elements and impurities found in the six candidate steels as well as in F82H [4]. The F82H 18 impurity list is measured from the 5 ton heat produced in Japan for JAERI by the NKK corporation. For 3Cr-3WV, RAAS, and 430-FS where the steel is relatively new or the impurity list and/or density are missing, the F82H impurities and/or density are used since doing physical property measurements on new steels may be too far into the future. The impurities in this table are labeled "nominal" impurities.

In an effort to reduce the long-term radioactivity, Klueh [4] provided a list of the lowest 17 impurities that have ever been measured in various steels. In other words, these are the lowest concentrations that have been achieved in large-scale melting and fabrication practices. They are not specific to any particular steel composition and should be achievable at present with a relatively modest effort and cost. This means such a list of controlled impurities would be achievable for any of the candidate steels for the VV (austenitic, ferritic, bainitic, or ferritic/martensitic). To quantify the impact on the activation, we examined the "nominal" impurities (given in Table 2) as well as the "present" impurities (given in Table 3) for all seven steels, including F82H.

Table 2. Compositions of Candidate Steels with "Nominal" Impurities
(in weight %; * indicates weight part per million (wppm))

Alloy	MF82H	316_SS	ORNL-FS	RAAS	3Cr_3WV_FS	430_FS	Mn_AS
Density (g/cm^3)	7.89	7.966	7.78	7.966	7.89	7.7	7.82
B		0.001					0.0025
C	0.1	0.0225	0.1	0.1	0.1	0.12	0.1
N		0.07					0.19
O		0.002					
Al	1.40E-03	0.05		1.40E-03	1.40E-03	1.40E-03	5.0E-02
Si		0.5	0.25	0.5	0.14	1	0.55
P		0.025					0.016
S		0.0075					0.008
K		*5					
Ti		0.15		0.5			
V	0.2	0.004	0.025		0.25		0.02
Cr	7.5	17.5	9	15	3	17	10.1
Mn		1.8	0.5	1.7	0.5	1	17.5
Fe	90.11586	64.938	88.073862	64.883258	92.945858	80.815858	71.2434
Co	*28	0.05	*34	*28	*28	*28	
Ni	*474	12.25	*402	15	*474	*474	0.1
Cu	*100	0.1		1.00E-02	*100	*100	*600
Y				0.3			
Zr		0.002					
Nb	*3.3	0.01	*4	*3.3	*3.3	*3.3	
Mo	*21.0	2.5	*70	*21	*21	*21	*600
Pd	*0.05		*0.18	*0.05	*0.05	*0.05	
Ag	*0.1		*0.16	*0.1	*0.1	*0.1	
Cd	*0.4		*0.05	*0.4	*0.4	*0.4	
Sn		0.002					
Ta	0.02	0.01	*0.07				
W	2	*10	2	2	3		
Os	*0.05		*0.02	*0.05	*0.05	*0.05	
Ir	*0.05		*0.05	*0.05	*0.05	*0.05	
Pb		*8					*1
Bi	*0.02	*8	*0.05	*0.02	*0.02	*0.02	
Eu	*0.05		*0.05	*0.05	*0.05	*0.05	
Tb	*0.02			*0.02	*0.02	*0.02	
Dy	*0.05		*0.05	*0.05	*0.05	*0.05	
Ho	*0.05		*0.05	*0.05	*0.05	*0.05	
Er	*0.05		*0.05	*0.05	*0.05	*0.05	
U	*0.05		*0.6	*0.05	*0.05	*0.05	

Table 3. Compositions of Candidate Steels with "Present" Impurities
(in weight %; * indicates wppm)

Alloy	MF82H	316_SS	ORNL-FS	RAAS	3Cr_3WV_FS	430_FS	Mn_AS
Density (g/cm^3)	7.89	7.966	7.78	7.966	7.89	7.7	7.82
B		0.001					
C	0.1	0.0225	0.1	0.1	0.1	0.12	0.1
N		0.07					0.19
O		0.002					
Al	*30	*30	*30	*30	*30	*30	*30
Si		0.5	0.25	0.5	0.14	1	0.55
P		0.025					0.016
S							0.008
Ti		0.15		0.5			
V	0.2	0.004	0.025		0.25		
Cr	7.5	17.5	9	15	3	17	10.1
Mn		1.8	0.5	1.7	0.5	1	17.5
Fe	90.173301	65.060601	88.118301	64.894601	93.003301	80.873301	71.529295
Co	*8	*8	*8	*8	*8	*8	*8
Ni	*13	12.25	*13	15	*13	*13	*13
Cu	*10	0.1	*10	*10	*10	*10	*10
Y				0.3			
Nb	*0.5	*0.5	*0.5	*0.5	*0.5	*0.5	*0.5
Mo	*5	2.5	*5	*5	*5	*5	*5
Pd	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05
Ag	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05
Cd	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05
Ta	0.02	0.01					
W	2	*10	2	2	3		
Os	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05
Ir	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05
Bi	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05
Eu	*0.02	*0.02	*0.02	*0.02	*0.02	*0.02	*0.02
Tb	*0.02	*0.02	*0.02	*0.02	*0.02	*0.02	*0.02
Dy	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05
Ho	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05
Er	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05	*0.05

5. Activation Model, Codes, and Methodology

The latest ARIES-CS design [2] in the ARIES series [14] was considered in this analysis as the ongoing ARIES-ACT tokamak design is still evolving. However, redoing analysis for the ARIES-ACT will not alter the main conclusions of this study. All ARIES designs deliver 1000 MW of net electric power. The reference radial build for ARIES-CS is shown in Fig. 1. The blanket and shield help protect the VV and all three components protect the superconducting magnets against radiation. The key activation-related parameters for ARIES-CS are the 2.6 MW/m^2 average neutron wall loading (NWL), 40 full power year (FPY) plant lifetime, and 85% overall system availability. Within the 28 cm thick VV, 28% by volume consists of the primary steel structure, 23% borated version of the primary steel (typically contains 3 wt% boron), and the remainder is the water coolant (on which no activation calculations were performed).

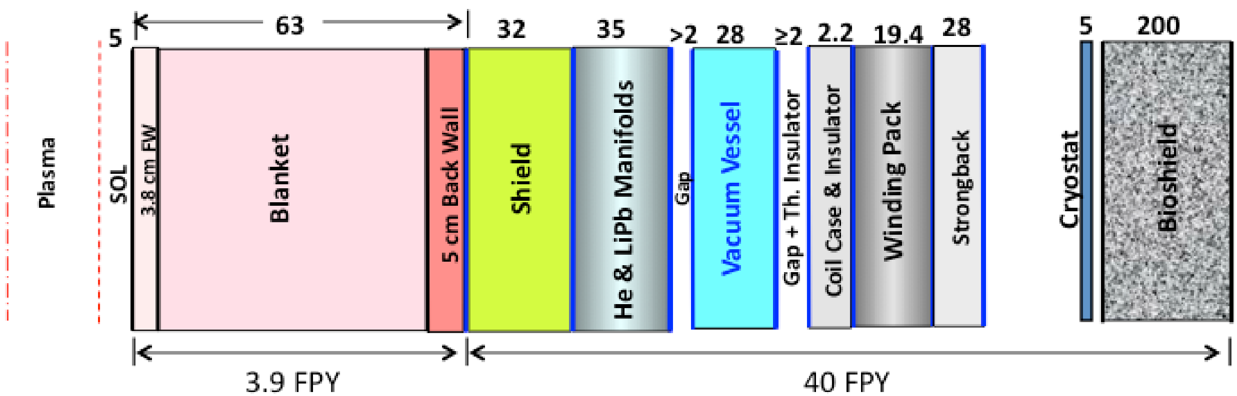

Figure 1. One-dimensional model of ARIES-CS radial build showing the replaceable (3.9 FPY) and permanent components (40 FPY).

The activation and environmental impact of the VV are the primary concern in this analysis. There is a growing international effort to avoid the geological disposal of radioactive materials. Instead, recycling (reuse within the nuclear industry) and clearance (release to the commercial market if materials contain traces of radioactivity) offer an alternate, more environmentally attractive means for dealing with the radwaste stream [3]. In this report, all three scenarios (recycling, clearance, and geological disposal) were investigated for a VV made out of the seven candidate steels with "nominal" impurities. In addition, the VV steels were reexamined with the list of "present" impurities.

In carrying out the analysis, the waste disposal rating (WDR) was calculated for a fully compacted waste using both Fetter's [15] and NRC [16] limits. These ratings determine whether the steel qualifies as low-level or high-level waste at a specific time after shutdown (typically 100 years). In addition, the NRC clearance standards [17] were used for calculating the clearance index [18].

Two codes were used to conduct the activation analysis. The first code is DANTSYS, a discrete ordinate, neutral particle transport code [19]. For each steel, the transport code was run using a one-dimensional cylindrical geometry with the FENDL-2.1 cross section data library [20] in 175 neutron group and 42 gamma group structure. DANTSYS generates the detailed radial distribution of the neutron flux throughout the radial build. The flux file couples with the second activation code ALARA [21,22] that uses the FENDL-2 activation library [23] with an operation schedule reflecting that of ARIES-CS (40 FPY with 85% availability). The results from the ALARA code give the recycling dose, CI, and WDR for each of the seven steels at various times after shutdown.

6. Activation Results

Using the activation information, recommendations can be made as to which materials would be best suited for the ARIES VV as considerations must be made to ensure that the VV is recyclable or clearable and classifies as LLW at the end of its service lifetime.

Firstly, we applied the recycling approach to the VV. It can potentially be recycled using advanced remote handling (RH) equipment. The variation with time of the recycling dose is shown in Fig. 2 for two steels: F82H and 316-SS. The other steels display similar behavior. For all steels, ^{54}Mn (from Fe) is the main contributor to the dose for up to 10 y. Storing the VV temporarily for several years helps drop the dose by a few orders of magnitude before recycling.

Secondly, we calculated the clearance index of the various steels. This index serves to identify which materials could be released to the commercial market to fabricate as consumer products. A clearance index (CI) can be computed as the weighted sum of all nuclide specific activities (in Bq/g) divided by the corresponding clearance limits. A material must have a CI $<$ 1 to be clearable. Even after an extended storage period of 100 y, none of the candidate VV steels are clearable, as shown in Fig. 3. The main contributors at 100 years are given in Table 4. The "present" impurities helped reduce the CI somewhat, but not enough to clear the VV at 100 years. According to Table 4, Eu-152, Tc-99, Nb-94, Co-60, and Ni-63 have the largest impact on the CI. If the amount of these radioisotopes and their precursors were reduced, it is possible that some of the steels could be made clearable.

Thirdly, we classified the VV for disposal based on its WDR. Note that the recycling and clearance are more environmentally attractive options [3] while the geological disposal is the least preferred option for any fusion component. Before a radioactive material can be disposed of, the level of its radioactivity must be classified. The two official classifications described by the Nuclear Regulatory Commission (NRC) are high-level waste (HLW) and low-level waste (LLW) [13]. One of the main goals of fusion is to produce no HLW in order to avoid deep geological burial. By definition, the WDR is the ratio of the specific activity (in Ci/m^3) to the allowable limit summed over all radioisotopes.

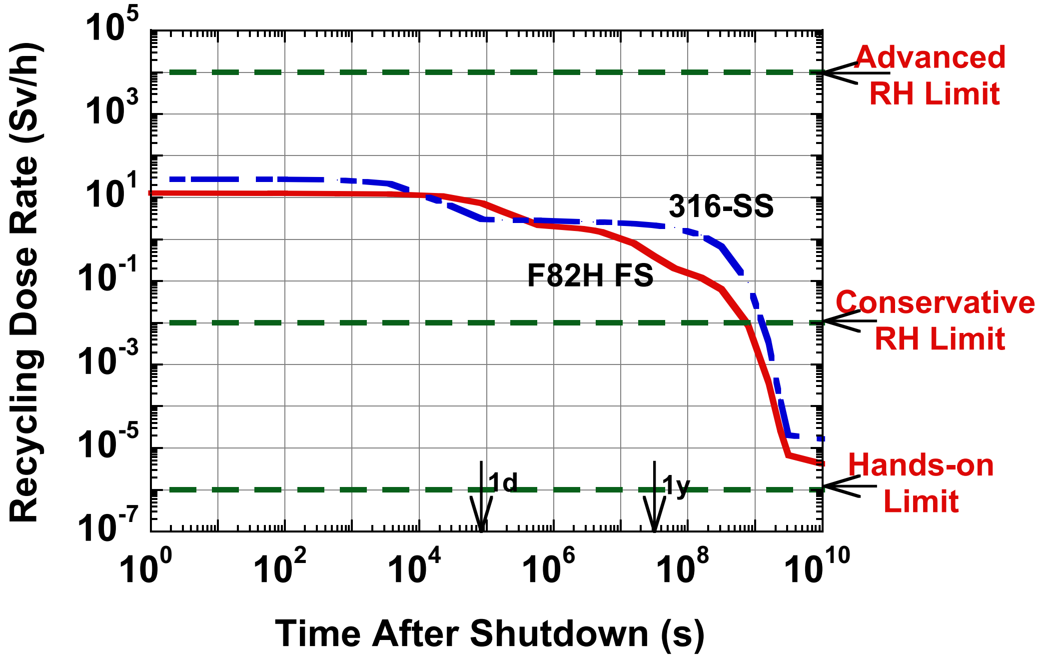

Figure 2. Variation of dose to RH equipment with time after shutdown for selected steels.

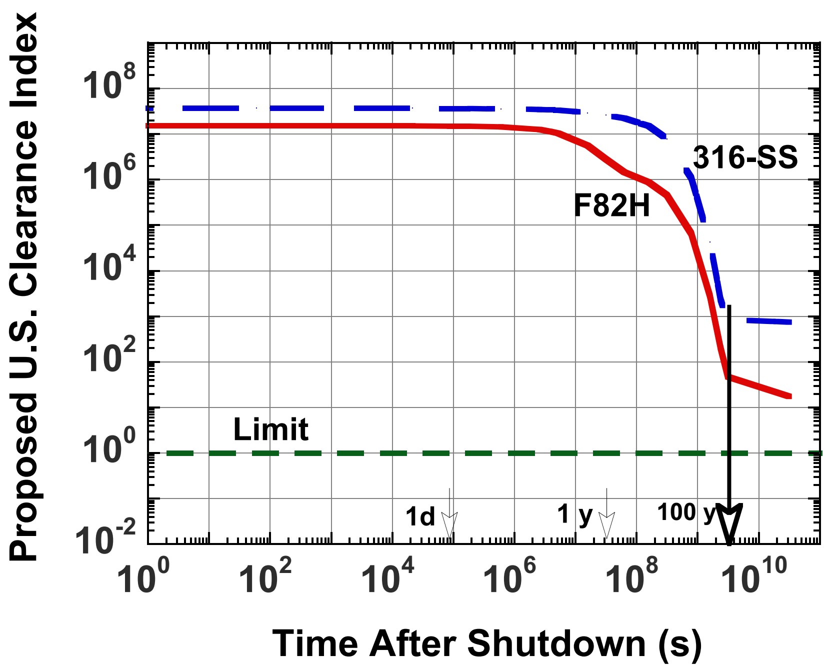

Figure 3. Decrease of clearance index with time after shutdown for selected VV steels.

Table 4. Clearance Index Values and Main Contributors at 100 years after Shutdown

	Clearance Index (Nominal)	Clearance Index (Present)
F82H	**47.1** Eu-152 (51.15%) Nb-94 (36.17%) Co-60 (7.36%)	**14.0** Eu-152 (68.83%) Nb-94 (18.44%) Co-60 (7.27%)
316_SS	**847.45** Nb-94 (59.08%) Tc-99 (22.90%) Mo-93 (8.69%) Co-60 (7.00%)	**300** Tc-99 (64.45%) Mo-93 (24.52%) Ni-63 (4.74%)
ORNL-FS	**58.3** Eu-152 (40.45%) Nb-94 (34.37%) Co-60 (6.98%)	**13.8** Eu-152 (68.57%) Nb-94 (18.61%) Co-60 (7.32%)
RAAS	**42.7** Eu-152 (55.18%) Nb-94 (11.72%) Ag-108m (2.50%)	**29.8** Ni-63 (54.92%) Eu-152 (29.66%) Nb-94 (8.33%) Co-60 (3.29%)
3Cr_3WV_FS	**45.1** Eu-152 (49.93%) Nb-94 (37.28%) Co-60 (7.49%)	**13.3** Eu-152 (67.82%) Nb-94 (19.18%) Co-60 (7.47%)
430_FS	**47.3** Eu-152 (51.51%) Nb-94 (35.68%) Co-60 (7.32%)	**14.1** Eu-152 (69.11%) Nb-94 (18.13%) Co-60 (7.19%)
Mn_AS	**18.4** C-14 (67.06%) Tc-99 (22.62%) Mo-93 (9.47%)	**25.3** C-14 (48.87%) Eu-152 (35.36%) Nb-94 (9.20%)

The NRC has defined three more categories for LLW: Class A, B, and C. Class A is the least hazardous type of waste. The NRC limits were issued for several radionuclides, most of which are fission products and transuranics with no relevance to fusion. In the early 1990s, Fetter used the NRC methodology to develop Class C limits for fusion-specific radionuclides. Because of uncertainties in the assumptions, a range of limits, rather than a single value, was assigned for some beta emitters. Here, the upper and lower ends of the range are referred to as the Fetter-Hi and Fetter-Lo. As expected, Fetter-Lo limits result in higher WDR compared to Fetter-Hi limits.

For the VV, the WDRs was evaluated for both sets of steels (with “nominal” and “present” impurities) at the end of the 100 y institutional monitoring period, using Fetter-Hi, Fetter-Lo, NRC-Class-C and NRC-Class-A limits. We first examined the Class C waste shown in Fig. 4. Here, the WDR was acquired using a combination of Fetter’s limits and NRC-Class-C limits. The highest value of the two evaluations is reported here for individual steels. As seen in Fig. 4, 316-SS must be excluded for generating HLW. All other steels fall well within the Class C LLW classification (WDR < 1). A reduction in the WDR of all steels, except 316-SS and RAAS, was notable when using “present” impurities. The main isotopic contributors to the WDRs can be found in Tables 5 and 6 for both “nominal” and “present” impurities, respectively.

Next, for the six steels with Class C WDR < 0.1, we identified which steel qualifies as Class A LLW; this designation is sought due to the cost savings encountered when disposing of materials in repositories. Figure 5 presents the Class A WDR, which was compiled using only NRC-Class-A limits; there are no counterpart limits by Fetter. From this figure, the RAAS disqualifies while all remaining steels qualify as Class A LLW. The 15 wt% Ni contributes to ~98% of the Class A WDR for RAAS. If less than 5 wt% Ni could be tolerated, RAAS can meet the Class A disposal requirement.

7. Recommended Steel for ARIES VV

The low-activation 3Cr-3WV steel is recommended for further consideration because of two significant advantages over the other candidates. Firstly, it qualifies as Class A LLW and therefore, meets the compositional requirements for long-term waste disposal without presenting serious concerns about decay heat. The main features of the 3Cr-3WV steel include superior weldability, radiation damage tolerance, adequate corrosion behavior at 20-300°C, and high toughness microstructure that develops during post-weld cooling (tempering probably not necessary). This steel may be sufficiently corrosion resistant and there is a real possibility that steels in this class could be tailored so that a weld microstructure with mechanical properties similar to the base metal could be generated during post-weld cooldown and obviate the need for PWHT. However, there is very limited data on radiation hardening and DBTT shifts, but it is unlikely that the hardening behavior would be much different from that of F82H. If this turns out to be the case, then the possible magnitude of DBTT shifts needs to be investigated, taking into account the section thickness, geometry of the VV structure, neutron dose, etc. If radiation-induced changes in tensile properties and fracture properties are unmanageable, the options of operating the VV at ~350^{o}C or the possibility of periodic annealing at 450-500^{o}C to recover the radiation damage will require more careful study. There is much interest in the 3Cr-WV steels for reactor vessels, pipelines and boiler tubes, not only because of their good strength and fracture toughness properties, but also because of the huge savings that would result from not requiring a PWHT.

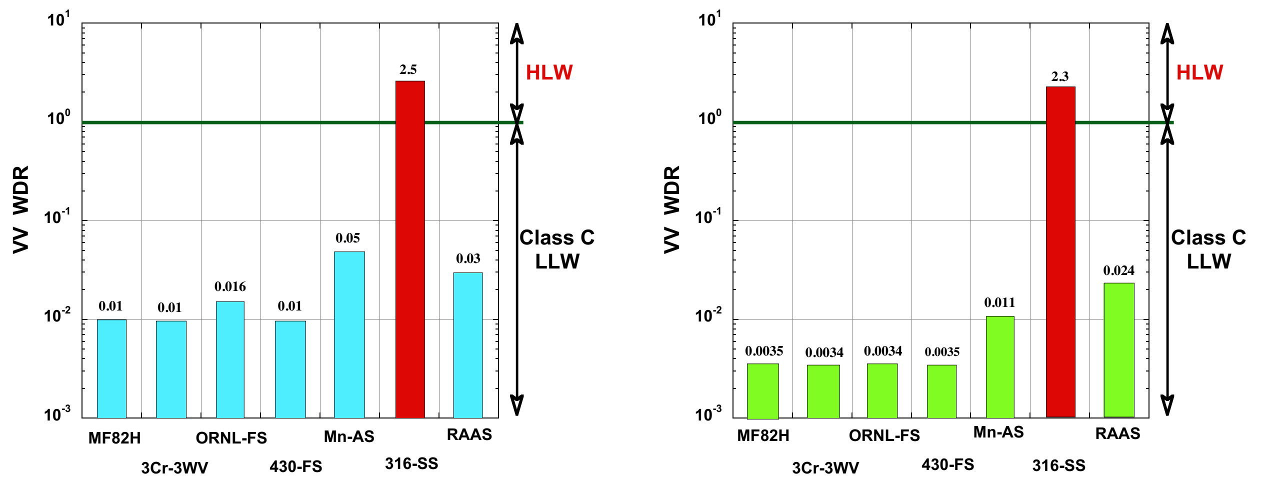

Figure 4. Class C WDR at 100 years after shutdown for the seven candidate VV steels with "nominal" impurities (left) and "present" impurities (right).

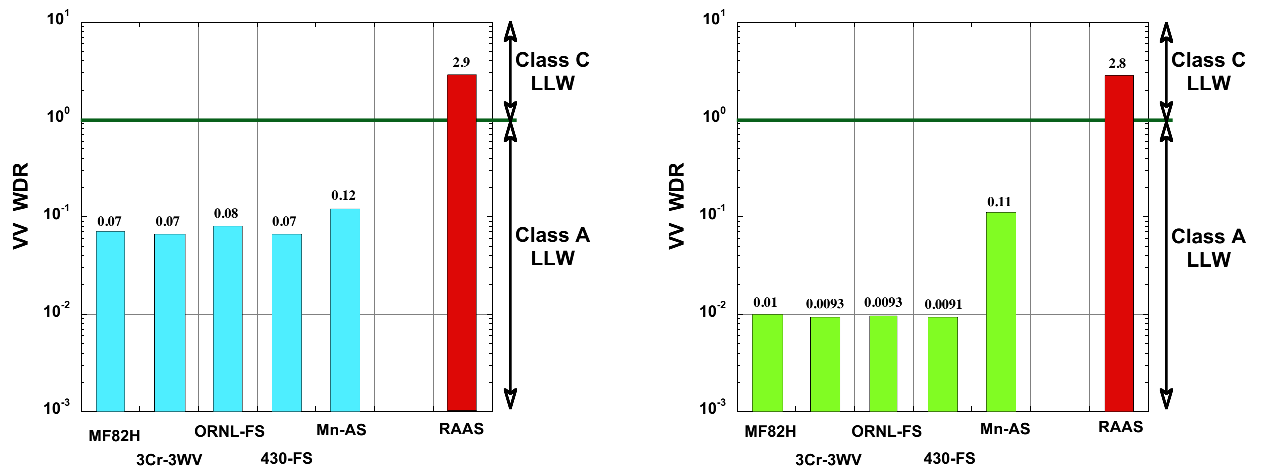

Figure 5. Class A WDR at 100 years after shutdown for the six VV steels that passed the Class C qualification with "nominal" impurities (left) and "present" impurities (right).

Table 5. Waste Disposal Ratings at 100 years after Shutdown ("Nominal" Impurities)

	Fetter-Hi	**Fetter-Lo**	**NRC-C**	**NRC-A**
F82H	**8.4e-03** Nb-94 (72.2%) Tc-99 (2.32%) Ho-166m (25.1%)	**1.0e-02** Nb-94 (59.75%) Tc-99 (19.19%) Ho-166m (20.76%)	**6.1e-03** Nb-94 (98.67%)	**7.0e-02** Nb-94 (86.27%) Ni-63 (13.22%) Ni-59 (0.45%)
316_SS	**0.4** Tc-99 (55.42%) Nb-94 (44.04%)	**2.5** Tc-99 (92.53%) Nb-94 (7.35%)	**2.1e-01** Nb-94 (86.64%) Ni-63 (5.57%)	**4.4** Ni-63 (54.12%) Nb-94 (42.09%) Ni-59 (1.83%) Tc-99 (1.06%)
ORNL-FS	**9.9e-03** Nb-94 (72.51%) Ho-166m (20.73%) Tc-99 (6.39%)	**1.6e-02** Nb-94 (46.05%) Tc-99 (40.55%) Ho-166m (13.16%)	**7.3e-03** Nb-94 (98.95%)	**8.0e-02** Nb-94 (90.02%) Ni-63 (9.50%)
RAAS	**1.0e-02** Nb-94 (56.81%) Ni-59 (21.82%) Ho-166m (19.41%)	**1.2e-02** Nb-94 (48.25%) Ni-59 (18.59%) Ho-166m (16.54%) Tc-99 (15.33%)	**2.9e-02** Ni-63 (47.32%) Ni-59 (32.22%) Nb-94 (20.44%)	**2.9** Ni-63 (94.73%) Ni-59 (3.23%) Nb-94 (2.05%)
3Cr_3WV_FS	**8.2e-03** Nb-94 (72.83%) Ho-166m (24.51%) Tc-99 (2.32%)	**9.9e-03** Nb-94 (60.23%) Ho-166m (20.27%) Tc-99 (19.22%)	**6.1e-03** Nb-94 (98.74%) Ni-63 (0.71%)	**6.9e-02** Nb-94 (87.00%) Ni-63 (12.51%) Ni-59 (0.43%)
430_FS	**8.2e-03** Nb-94 (71.45%) Ho-166m (25.91%)	**9.9e-03** Nb-94 (59.21%) Ho-166m (21.47%) Tc-99 (19.03%)	**5.9e-03** Nb-94 (98.63%)	**6.8e-02** Nb-94 (85.91%) Ni-63 (13.57%)
Mn_AS	**4.9e-03** Tc-99 (96.88%) C-14 (2.69%)	**4.9e-02** Tc-99 (97.25%) C-14 (2.70%)	**1.0e-02** C-14 (97.56%)	**1.2e-01** C-14 (83.37%) Ni-63 (15.31%)

Table 6. Waste Disposal Ratings at 100 years after Shutdown ("Present" Impurities)

	Fetter-Hi	**Fetter-Lo**	**NRC-C**	**NRC-A**
MF82H	**3.1e-03** Ho-166m (68.30%) Nb-94 (29.78%)	**3.5e-03** Ho-166m (60.16%) Nb-94 (26.23%) Tc-99 (13.24%)	**9.2e-04** Nb-94 (99.65%)	**9.4e-03** Nb-94 (97.07%) Ni-63 (2.69%)
316_SS	**2.3e-01** Tc-99 (97.69%)	**2.3** Tc-99 (99.74%)	**2.9e-02** Ni-63 (40.59%) Ni-59 (27.49%) Tc-99 (15.41%) C-14 (13.35%)	**2.5** Ni-63 (93.18%) Ni-59 (3.16%)
ORNL-FS	**3.0e-03** Ho-166m (68.24%) Nb-94 (29.84%)	**3.4e-03** Ho-166m (60.12 %) Nb-94 (26.29 %) Tc-99 (13.23 %)	**9.0e-04** Nb-94 (99.65%)	**9.3e-03** Nb-94 (97.11%) Ni-63 (2.65%)
RAAS	**5.2e-03** Ni-59 (43.22%) Ho-166m (38.45%) Nb-94 (16.99%)	**5.8e-03** Ni-59 (39.30%) Ho-166m (34.96%) Nb-94 (15.45%) Tc-99 (7.71%)	**2.4e-02** Ni-63 (57.25%) Ni-59 (38.99%)	**2.8** Ni-63 (96.40%) Ni-59 (3.28%)
3Cr_3WV_FS	**3.0e-03** Ho-166m (67.62%) Nb-94 (30.44%)	**3.4e-03** Ho-166m (59.45%) Nb-94 (26.76%) Tc-99 (13.42%)	**9.1e-04** Nb-94 (99.66%)	**9.3e-03** Nb-94 (97.23%) Ni-63 (2.53%)
430_FS	**3.1e-03** Ho-166m (69.22%) Nb-94 (28.93%)	**3.5e-03** Ho-166m (61.17%) Nb-94 (25.56%) Tc-99 (12.91%)	**8.9e-04** Nb-94 (99.64%)	**9.1e-03** Nb-94 (96.98%) Ni-63 (2.77%)
Mn_AS	**3.0e-03** Ho-166m (66.52%) Nb-94 (27.36%)	**4.6e-03** Ho-166m (43.90%) C-14 (29.08%) Nb-94 (18.05%)	**1.1e-02** C-14 (92.33%) Nb-94 (7.64%)	**1.1e-01** C-14 (92.13%) Nb-94 (7.63%)

8. Conclusions

Seven different types of steel were examined in order to determine which steel would be the best candidate for the ARIES vacuum vessel. The main concerns are related to activation (recyclable, clearable, LLW or HLW), properties under irradiation (DBTT shift, corrosion resistance, swelling, etc.) and fabrication of a sizable VV (welding, PWHT, etc.). Although none of the candidate steels were clearable under ARIES operating conditions, all steels are recyclable and qualify as Class A LLW with either "nominal" or "present" impurities, except 316-SS and RAAS.

The most popular F82H FS for Demo and power plants has the advantage of having a composition specifically tailored to generate only LLW. However, it was initially envisaged as a FW/blanket structural material operating above 350^{o}C. As a VV material, it would be operating at 150-200^{o}C and subject to shifts in the DBTT. In addition, there is also the necessity for a carefully controlled PWHT at ~750^{o}C after assembly, welding, and rewelding. For these reasons, the F82H FS is not suitable for the ARIES VV. It appears from this study the newly developed 3Cr-3WV bainitic FS mitigates most of the identified F82H problems and would be a good choice for the ARIES VV. In fact, this steel meets the activation requirements and has the potential to satisfy the fabrication requirements for the ARIES VV. It is recommended for further consideration because of several advantages over other candidate steels:

- Classifies as Class A LLW with either "nominal" or "present" impurities
- Generates low decay heat
- Could satisfy fabrication requirements
- Probably satisfies strength, fracture toughness, and ductility requirements, but still needs low temperature irradiation data on hardening/shifts
- Need to assess IASCC (irradiation assistant stress corrosion cracking), but is being developed for high temperature water/steam boiler applications.

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