Development of Nanosized Carbide Dispersed Advanced Radiation Resistant Austenitic Stainless Steel (ARES) for Generation IV Systems
Mr. Ji Ho Shin, KAIST, Korea (Public Vote Winner of the 2021 Pitch Your Gen IV Research Competition)

Berta Oates
Doing the introduction today is Dr. Patricia Paviet. Dr. Paviet is the Group Leader of the Radiological Materials Group at Pacific Northwest National Laboratory. She is also the Chair of the Gen IV International Forum, Education and Training Working Group. Patricia?

Patricia Paviet
Thank you so much, Berta. Good morning or good evening, everyone. We are happy to have with us today, Dr. Ji Ho Shin, who was the Popular Vote Winner of the 2021 Pitch Your Gen IV Research competition.

He recently completed his Ph.D. at the Korea Advanced Institute of Science and Technology in the field of nuclear materials on the subject of "Development of Nano Carbide Dispersed Advanced Radiation Resistant Austenitic Stainless Steels for Reactor Internals". His Ph.D. focuses on the development of next-generation nuclear in-core materials, including Small Modular Reactor, Sodium Fast Reactor, and fusion reactor to demonstrate the superior radiation resistant feature.

He is currently a post-doctoral fellow in the Korea Atomic Energy Research Institute.

Without any delay, Ji Ho, I give you the floor. I thank you again for giving us your webinar. Thank you.

Ji Ho Shin
Hello, everyone. My name is Ji Ho Shin, presenting on "Development of Nanosized Carbide Dispersed Advanced Radiation Resistant Austenitic Stainless Steel for Generation IV Systems".

Before the introduction of my research, I have to say that I am really thankful to give me an opportunity to present my research in this Generation IV International Forum.

Structural materials represent the key for containment of nuclear fuel and efficient products, as well as reliable and thermodynamically efficient production of electrical energy from nuclear reactors. As the new concept of reactors has developed to guarantee the life extension of existing reactors and development of next-generation reactors, design of high-performance radiation-resistant material is considered as a key strategy.

Among the various Gen IV reactors, this page shows six Generation IV concepts, which are sodium fast reactor, lead fast reactor, gas fast reactor, very high temperature reactor, supercritical water-cooled reactor, and molten salt reactor.

This representative retro type has a propulsive operating temperature and lifetime displacement damage levels for structural materials. Especially SFR would be expected around 200 dpa, which is similar with MSR type.

In addition, depending on operating temperature regime, where they can be utilized in a neutron irradiation environment, existing structural materials have rather remitted, as shown in bottom side figure. At low temperature, the reduced ductility associated with a low-temperature radiation hardening creates condition where fracture continues to reduce by low temperature irradiation. Therefore, the operating temperature is restricted to higher temperature where embrittlement does not occur for anticipating normal and transient operating conditions.

Furthermore, the upper operating temperature remits is typically determined by thermal creep strength or high-temperature helium embrittlement consideration.

In addition to temperature effect, the microstructural evolution in the irradiated material depending on the material parameter such as primary knock-on atom energy, displacement dose, damage rate, crystal structure, solute, and transmutant elements which are caused by energetic neutron irradiation. This figure shows example of typical temperature-dependent TM microstructures produced by irradiated materials.

At low temperature, stacking fault tetrahedral, dislocation loops are observed, and network dislocations, void, bubbles, precipitates, and solute segregation are observed at intermediate temperature. At high temperature, helium embrittlement can be caused due to the grain boundary helium cavities. Thus, depending on temperature, evolution of defects can be differed from dislocation loop to void and helium cavities.

More specifically, radiation damage caused five main threats to the operation of structural materials emerging at different operating temperature and damage levels. Coupled with irradiation temperature and damage level, radiation hardening, and embrittlement faced instability, irradiation creep, void swelling, and helium embrittlement would happen.

Meanwhile, Generation IV system requires these four primary areas, and the materials should be developed for the various types of Generation IV vectors as follows: The expected operation temperature, neutron spectrum, and coolant compatibility, as you can see on the right-hand side table.

Principle challenges in the development of Gen IV nuclear reactor is the discovery of all advanced structural material that can endure extreme environments. Indeed, our Gen IV reactors concept place a very high burden on core materials which will have to withstand high operating temperature, intense, fast, neutron flux, and contact with the corrosive environments.

Furthermore, the objective of the Gen IV program includes extended design lifetime to 60 years, increased fuel burnup and cycle length as compared to current reactors. As a result of that, the maximum radiation dose for in-core materials could exceed 200 dpa.

Currently, ferritic-martensitic steels are being considered as a candidate for internal materials comparing with austenitic stainless steels because of their lower thermal expansion, higher thermal conductivity, and higher radiation tolerance, etcetera. In particular, the swelling resistance of FM steel is much better than austenitic stainless steel in general. However, this material has been known to be susceptible to DBTT, which caused the low fracture toughness and fail by brittle fracture at room temperature. Moreover, most FM steels have lower high-temperature oxidation resistance compared to austenitic stainless steels.

Meanwhile, austenitic stainless steels also have been extensively used in the past as the first candidate materials of sodium fast reactor hexagonal wrapper tubes. However, as you can see, the poor swelling resistance is one of the major reasons why an austenitic stainless steel was remitted in the Gen IV system compared to FMS or FM-ODS. Therefore, my research strategy regarding the alloy development has increased the poor swelling resistance based on the austenitic stainless steels.

Here is the content for today's presentation. First, I will introduce the development of the ARES alloy for Gen IV rector. Second, characteristic of radiation resistance for the developed alloy. First topic is development to ARES alloy, which full name is Advanced Radiation Resistant Austenitic Stainless Steels.

Up to now, there are lots of development strategies to increase the radiation resistant characteristics by using an alloy element such as high nickel, low silicon content, high coincident site lattice fraction, high Schmid factor, low Taylor factor, small grain, cold working, and the precipitates, etcetera.

As you can see the right bottom side figure, depending on the chemical composition and microstructural features, void swelling resistance increased as following the damage level.

In addition, as mentioned in previous page, three general strategies can be employed to increase radiation tolerance in materials: Radiation resistance matrix phase, immobilize point effect, and engineered high sink strength microstructures. Overall, the strategies are based on the introduction of various microstructural features, which may act as a sink site or point defect.

For example, as you can see, nano-grain, which have 100 nanometer square grain, dislocation coupled with the precipitate FMS or FM-ODS, high-entropy alloy, and twin boundary. The sinks can promote a recombination of radiation in this point defect, which are the vacancies and the interstitials, and thereby delay the formation of the void and dislocation loops, which would be caused by the clustering of the vacancies and interstitials. The effect of these microstructural features has been demonstrated by calculated theory models and lots of the experimental studies in suppressing void swelling or reducing radiation-induced element segregation.

However, forming the nanoscale grain by severe plastic deformation technique by using equal channel angular pressing will be limited in terms of their applicability to reacting into the components, especially those with the complex shape and large size. Also, powder metallurgy, coupled with mechanical alloying or hot isostatic pressing has the same result with the ECAP method.

Another thing is cold rolling and precipitation heat treatment for forming a precipitate inside the grain. However, as you can see bottom side figure, stability from the grains were formed and the precipitate can be formed along the deformation band along the rolling direction, which have stringer-type precipitates.

Therefore, in this study, we developed a new approach to form a high density of the uniformly distributed nanosized carbide in an austenitic stainless-steel matrix. For this, we developed a model alloy and applicable thermomechanical processing to control the dislocation structure, microstructure, and precipitation of fine carbide.

As you can see this flow chart, to develop the ARES alloy, the chemical composition was defined based on the thermodynamic simulation, and the ingots were made by VIM method, followed by a newly designed thermomechanical processing. The detailed design procedure will be presented in the next page. After that, microstructure analysis in terms of the grain structure and precipitation morphologies of precipitates, mechanical, corrosion test, and calculation of the sink strengths were conducted.

Among the alloys, irradiation and SCC resistance were conducted for the candidate alloys which were selected from the screening process. In this study, only its third-phase alloy group will be presented.

The proposal of the alloy design was to identify an appropriate set of the composition range from austenitic stainless steels for nuclear power plant application with improved swelling resistance. The minor elements were controlled to form a nanosized precipitate. Previously, minor elements such as niobium, vanadium, and titanium have been extensively used to form a carbide and nitride and steel since they have formed a stable precipitate based on the enthalpies and the solubility.

Meanwhile, generally, nitride can form a very high stable precipitate, and the carbon nitride during the thermomechanical processing. Therefore, niobium addition was intended to form a large amount of the uniformly distributed niobium carbide precipitate within the austenite matrix as a primary precipitate compared to the titanium nitride.

Here is briefly introduction of all the design process conducted by Thermo-Calc software. The composition is only regarding the third phase alloys. To form nanosized precipitates, titanium and nitrogen, and niobium and carbon contents were set based on the thermodynamics.

Moreover, to prevent the calcium precipitate, small amount of the manganese and molybdenum were controlled. For sufficient corrosion resistance in the anticipated operating environments, the chromium content was selected as 24 weight percent, and the nickel competition was selected as 21 weight percent to form a fully austenitic phase by balancing the high chromium contents.

Consequently, the simulation results were shown in left side figure. Overall, the matrix is austenite phase, and a small amount of niobium carbide or titanium nitride were formed inside the grain.

All of the chemical composition for each batch are tabulated in this page. From the batch number 1, which is ARES-1 to ARES-8, the chemical composition was modified based on the microstructure and mechanical process. The ARES-6, ARES-7, and ARES-8 were belonging in Phase 3, batch; and high chromium, high nickel, and controlled minor elements are as shown in this table. A detailed information is published in previous patents.

To form a nanosized precipitate, newly designed thermomechanical processing should be needed. Right-hand side figure shows the schematic of the TMP applied to the plate of the ARES alloys considering the non-recrystallization temperature. The temperature concept was motivated from the thermomechanical processing of the low volatile [ph] steel, as you can see on the left-hand side.

The plates were homogenized at 1,200 Celsius for 2 hours and are cooled to the starting rolling temperature. Then, a total of the six rolling passes were applied to achieve around 70% thickness reduction after the final rolling temperature was reached.

Three different rolling conditions were applied according to the T_{NR}, non-recrystallization temperature, which was determined by a double-hit deformation test. Under the rolling condition of the 1HR, all six rolling passes were above the T_{NR}; for 2HR, four rolling passes were above the T_{NR} and two were below the T_{NR}; for 3HR, two rolling passes were above the T_{NR}, and rest of the passes were below the T_{NR}. Thus, from the 1 HR to 3HR, the number of the rolling passes below the T_{NR} gradually increased, as you can see in the table.

The detail regarding the rolling conditions is presented on the right bottom side of the table.

The plates were held for 2 hours for precipitation heat treatment followed by air-cooling to room temperature for the completion of the TMP.

The microstructure of the ARES alloy after the homogenizing heat treatment and before deformation is presented in OM image. The grains are large owing to the grain growth during the homogenization. However, because of the peeling effect of the titanium nitride, the grains had a mean size of 115 micrometers, which is almost half of that of the commercial 304 stainless steel with the same heat treatment condition. Thus, the grain growth was inhibited.

The microstructure of the hot-rolled and the crunched samples are shown in right upper figures. As the hot-rolling processes are conducted under the non-recrystallization temperature, the amount of the dislocations increased, but oversaturation dislocation and cell structure were observed in 3HR condition.

After precipitation heat treatment, only 2P condition shows uniformly distributed nanosized precipitation inside the grain. Meanwhile, 1P condition shows chrome-carbide and niobium carbide along the grain boundary, and 3P condition shows stringer-type precipitates on the dislocation pile-up region. Therefore, 2P condition was selected as a final TMP condition for this developed alloy.

The detailed microstructure regarding the nanosized niobium carbide precipitate is present in here. As you can see the BFTEM and HRTEM images, a large amount of the uniformly distributed nanosized niobium carbide precipitates are formed inside the grain, and the precipitation have cube-on-cube orientation relationship with offset matrix. Moreover, the precipitates have semi-horizontal relationship with matrix. Therefore, it can be said that stable niobium carbide precipitates were formed in the grain, in the matrix during the thermomechanical processing. In other batches, the microstructure features are dissimilar.

The site and the distribution of the nanosized precipitates are shown in this page. However, the nanosized niobium carbide precipitates have 5 to 10 nanometer as mean diameter with a volumetric density of the 10^{22} to 10^{23} per cubic meter. Especially the ARES 8 shows similar microstructure feature with well-known ODS alloys.

A new approach for forming a high density of the uniformly distributed nanosized carbide in an austenitic stainless-steel matrix was developed. To guarantee the corrosion properties, high chromium and high nickel contents are included, and minor elements were used to form nanosized precipitates. Compared to the commercial stainless steel and other radiation-resistant alloy, ARES alloy shows higher precipitation density and comparable to ODS alloy.

The second topic is "Radiation Resistance of ARES Alloy".

In this section, the radiation resistance characteristic was focused on void swelling and radiation hardening resistance. As I briefly explained the effect of the defect sink on the radiation resistance at introduction parts, they may act as a sink site for point defects. The sinks can promote a recombination of radiation-induced point defect for vacancies and interstitials, and thereby delay the formation of the voids and dislocation loops, which would be caused by the clustering of them.

In addition to defect sinks, the major elements can relate with the radiation resistance. As you can see these figures, the beneficiary effect of the nickel between 15% and 60%, and conversely, the harmful effect of the chrome between 7% to 30% on the resistance of the swelling of ternary alloy by a strong ion irradiation dose.

It has been explained that the mobility of the vacancy depending on the binding energy of solid vacancy complexes, which can be formed more easily when the binding energy is greater than 0.2 to 0.3 electron volt. It has the recombination process. Especially nickel can increase the incubation dose for any initiation of the void swelling because it has high binding energy with vacancy, which is 0.26 electron volt. Meanwhile, addition of the chromium beyond 15% to 80% is not beneficial for swelling due to the low binding energy between the chromium and the vacancies, which is 0.06 electron volt.

The qualitative analysis results regarding the sink strength of the ARES alloy are inserted in the right-hand side figure to compare to radiation-resistant alloy, such as FMS and ODS alloys. Based on the calculated sink strength caused by the microstructure, the ARES alloy shows similar with FMS and all the stage of the ODS alloy are depending on the batches.

Here is the objective to evaluate the radiation resistance of the ARES alloy. The radiation experiments were conducted using an ion irradiation to emulate neutron irradiation. Then, measurement of the void size and density and the calculation of void swelling were conducted in terms of the void swelling resistance.

In addition, to measure the irradiation hardening resistance, nano-indentation experiments were conducted before and after irradiation. Finally, radiation-resistant characteristic of the ARES alloys was compared to commercial austenite stainless steel.

First of all, to conform the beneficiary effects of the fine niobium carbide precipitate on swelling resistance, the swelling resistance of the newly developed ARES-6 alloy was investigated depending on the thermomechanical processing. The reference alloy was selected as 316 stainless steel. Both chemical compositions are tabulated in here, and the thermomechanical processing of each alloy is also tabulated.

Heavy ion irradiation was performed using 1.7-megavolt Tandem ion accelerator at Cambridge Laboratory for Accelerator-based Surface Science of the MIT.

The energy of the nickel ions used for the irradiation was 5 megaelectron volt, and the irradiation temperature was set to 500 Celsius in order to compensate a dose rate effect on microchemistry. A static, defocused beam was used to minimize the effect of the strong void suspension caused by the beam restrain.

Based on SRIM software, the depth profile of the radiation damage and dose rates caused by the nickel ions were calculated and plotted in the left-hand side figure. In this study, the voids were quantitatively analyzed at 100 nanometer in towards [ph] from the 400 to 800 nanometer depths in width of the plus/minus 50 nanometer.

We measured the void size and density. More than 50 high magnitude TEM images were taken over the irradiation area, and voids were measured by an under-focused condition. The voids volume was calculated according to the following equation. The overall microstructure of irradiation 316 stainless steels ARES alloys is presented in the right-hand side figure.

The calculated voids swelling based on the void size and density are plotted in the right-hand side figure. As is shown in the figure, void swelling at 8.5 dpa, which is correlated in 600-nanometer region from the surface was calculated as a 4.1% for ARES #6 HR and 4.7% for ARES #6 SA, respectively, from the linearly fit trend line.

Meanwhile, obviously, swelling was the smallest while the swelling resistance was the largest in the ARES-6P containing a large amount of the sink site, such as dislocations and precipitates. It can be said that even though the major elements in ARES alloy can effect on void swelling resistance when it comes to compare between the 316 stainless steel and ARES #6 assay condition, a large amount of the nanosized niobium carbide precipitates or the precipitates coupled with dislocations would present dominant factor to have suppressed the void with nucleation and growth, resulting in less void swelling in ARES-6P. Meanwhile, the dislocation itself in ARES #6HR would not provide effective sink sites.

However, microstructural instability caused by extreme irradiation condition conducted by 10 to the minus 3 dpa per second was observed. When you look at the BFTEM images and niobium carbide mapping data before and after ion irradiation, relatively larger niobium carbides were formed irregularly in the matrix differed from the initial microstructure feature. In addition, from the specific blue box which magnify the BFTEM, the boundaries of the nanosized niobium carbide in HRTEM images were not clear at all in Fourier-filtered images.

Essentially, reports deeper explained that depending on the dose rates, nanosized alpha-prime precipitates dissolved in the matrix. Dose rates 1.8 multiplied 10^{-3} dpa per second which was used in this study was too high to maintain the nanosized niobium carbide precipitates and caused less than expected improvement in swelling resistance for ARES-6P. Nonetheless, it is anticipated that the nanosized niobium carbide precipitates will be much stable at lower dose rate and the swelling resistance of the ARES-6P will be much greater in typical nuclear power plant conditions.

To verify the radiation resistance up to high damage level, additional heavy ion irradiations using 5 megaelectron volt Fe ions were performed with 1.7 megavolt tandem accelerator at Texas A&M Ion Beam Lab.

Irradiation temperature was set to 500 and 575 Celsius, considering the trade-off between the temperature and dose rates. Targeted damage was 200 dpa with a dose rate of the 5 multiplied 10^{-4} dpa per second at a 600-nanometer depth from the surface considered as the plateau region. After the heavy ion radiation, the overall microstructural features are showing below. Obviously, swelling was smaller in ARES alloy.

The representative BFTEM images are shown here to quantify the voids. The size and number density were measured for the regions from 400 to 800 nanometer to avoid measurement errors, as explained in the previous section. Based on the void size and number density, void swelling was calculated and plotted in (c) and (f) at 100 nanometer in total.

From the linearly fitted trend line in (c) and (f), void swelling of ARES-6P is around 3.1% and 2.2% at the 600-nanometer region, which are about one-eighth and one-sixth of those of the 316 stainless steel, respectively.

It has been known that nanosized precipitates can suppress void swelling by observing the radiation-induced vacancies and interstitials. Especially in the previous study, we showed that nanosized niobium carbide precipitates our dominant factor to improve the void swelling resistance of the ARES-6P. In this study, we showed that ARES #6P exhibits superior void swelling resistance, even in high damage level, which is 200 dpa compared to that of 12 chromium, the ferritic magnetic steel, that has been considered as one of polymerized material in next-generation nuclear systems.

Irradiation resistance, it is crucial when the nanosized niobium carbide precipitates, which acts as a sink site for the point defect, can be maintained under the irradiation. In order to analyze the precipitates, the representative BFTEM images and the niobium maps were used for the ARES-6P irradiates at 500 and 575 Celsius, respectfully.

Compared to the initial size and number density, precipitates are quite similar for ARES-6P irradiates at 575 Celsius, while there are smaller but more numerous for the ARES-6P irradiates at 500 Celsius.

To characterize niobium carbide precipitate in the irradiated ARES-6P, the orientation relationship between the matrix and niobium precipitates was analyzed using the high-resolution TEM images from the certain [ph] area from the BF images, which is taken from the 600-nanometer region. However, the cube-on-cube orientation relationship between the niobium carbide precipitates and the matrix were observed but observed reflections of the niobium carbide precipitates are different depending on where they are away from or nearby the void.

For the niobium carbide precipitates away from the void in both irradiated conditions, all reflections of niobium carbide precipitates were observed blurrily in FFT patterns, though the phase boundary between the niobium carbide precipitates and the matrix in HRTEM images were not clearly defined unlike the initial microstructure feature.

Contrary to the niobium carbide precipitates observed away from the void, the reflections of the niobium carbide precipitates formed near the void present only along the (111) plane. It can be said that niobium carbide precipitates would reprecipitate with the recalled niobium and the carbon atoms during the irradiation on the void swell phase. The size of the precipitate was smaller than the pre-existing precipitates. Nonetheless, the size and density of the precipitation were similar with initial microstructural features.

Comparing to typical radiation resistant alloys such as HT9, the cross-sectional void swelling microstructure was similar, as you can see on the left-hand side figure. Moreover, compared to commercial 304 stainless steel, cold-worked 316 stainless steel and other fulminating alloy, such as D9 and A709, the void swelling resistance of the ARES alloys shows dramatically enhanced feature, as you can see on the right-hand side figure.

The nanosized precipitates are generally considered as the neutral defect sinks for trapping and annihilating radiation-induced defects. The primary mechanism of the inhibition of void swelling is dramatic evolution of the radiation-induced defects along the precipitate-matrix interfaces at elevated temperature.

As you can see in the left side figure, some of the voids are attached along the interface, as you can see the HRTEM – HRSTEM image. As long as the defects are preferentially [ph] absorbed by a large amount of the nanosized precipitate, the creation of the lattice site does not lead to an increase of the total volume leaving a huge cluster, consequently giving a material good swelling resistance.

To measure the mechanical properties on the local area subjected to ion irradiation, nanoindentation was conducted. From nanoindentation regards [ph], the scale of bulk hardness values was measured by extrapolated values obtained based on the Nix-Gao model to eliminate the indentation size effect that is shown in left hand side figures. The bulk hardness value for the 316 stainless steels and ARES-6P plot in the right-hand side figure are tabulated.

Although ARES-6P shows somewhat greater hardness value than 316 stainless steel in un-irradiated condition, ARES-6P shows significantly lower hardness value than 316 stainless steel after irradiation. Or the increase in hardness value after irradiation are much smaller for ARES-6P and 316 stainless steel. Overall, ARES-6P shows enhanced irradiation hardening resistance based on the hardness difference and the post-irradiated hardness value compared to the 316 stainless steel. In addition, smaller increments of irradiation hardening at 575 Celsius compared to 500 Celsius is consistent with the void swelling result of the ARES-6P.

In this study, the newly developed ARES-6P exhibits superior void swelling resistance compared to 316 stainless steel under low and high damage level at high temperature. Significant dissolution and reprecipitation of niobium carbide precipitates were observed near voids for the ARES-6P irradiated at 500 Celsius. Dissolution of niobium carbide precipitates was less significant for the ARES-6P irradiated at 575 Celsius, which is suggesting the better stability of niobium carbide precipitates. The primary mechanism of the inhibition of the void swelling is dynamic evolution of radiation-induced defects along the precipitate's matrix interface at elevated temperature.

In addition, the degree of irradiation hardening measured by nano-indentation was much smaller, which can be correlated with less radiation-induced defects for ARES-6P than 316 stainless steel. Both significantly less void swelling and less irradiation hardening indicates the superior irradiation resistance of ARES-6P for the application of the next-generation nuclear system.

Similar with Gen IV systems, the fusion blanket structural materials will be subject to high heat load and high flux around 14 megaelectron volt fusion neutrons, and will experience internal pressure load, thermal stress, structural [ph] magnetic force in case of the plasma disruptions, and high-pressure load. In designing of the fusion reactor, selection of structural materials is very crucial with respect to power plant efficiency and economy competitiveness.

The typical requirements of fusion reactor blanket material are low activation, radiation resistance of up to 200 dpa, high temperature properties and long-term thermal stability, and productivity for mass production. Based on these requirements, the initial proposal of the ARES-F alloys design was to search for appropriate set of the composition range for nickel, manganese, and chromium, replacing or reducing the leaked contents to apply to fusion reactor. Therefore, as a first attempt to determine the major elements of alloy to use the base composition, thermodynamic calculation was performed using a commercial Thermo-Calc Software with the low steel set to TCFE9 in the database.

As a result, we have set up four types of alloys depending on the major elements and ARES-F number 3 shows the optimized alloy having a large number of tantalum carbides in homogenous grain structure, as you can see in the right-hand side figure. Similarly, which I presented regarding the ARES-6P, this ARES-F alloy, which for the fusion reactor, has a similar microstructure between them.

The preliminary result of the heavy ion irradiates ARES-F number 3 shows a good void swelling resistance, and its characteristic is similar with a typical FMS, such as BN-10 and 12 chromium FMS.

The ARES upgrade will be more optimized in the future in terms of the high temperature property and the helium embrittlement resistance.

Thank you for your attention.

Berta Oates
Thank you. If you have questions, do go ahead, and type them into the chat pod now. While questions are coming in, we'll take a quick look at the upcoming webinar presentations.

In June, a presentation on Nuclear Waste Management Strategy for Molten Salt Reactor Systems.

In July, A Gas Cherenkov Muon Spectrometer for Nuclear Security Applications.

In August, China's Multi-purpose SMR-ACP100 Design and Project Progress.

Let's see here.

I have seen some accolades. Thanks, Isabella, for that. It was an excellent presentation. I think it's apparent from the technical discussion that the next generation in our nuclear workforce is in good hands. I really appreciate Dr. Ji Ho Shin's expertise in his time that he shared with us.

I don't see – here we go.

Often, the critique is that ion irradiation does not fully simulate the nuclear irradiation behavior. Can you comment on this?

Ji Ho Shin
Could you speak again, please?

Berta Oates
Often, the critique is that ion irradiation does not fully simulate the nuclear irradiation behavior. Can you comment on that?

Ji Ho Shin
Yes. Generally, the ion irradiation is only used to emulate neutron irradiation because this has lots of advantage to use an ion irradiation. First of all, it's time-consuming to emulate a nuclear irradiation, but the big problem is the totally difference regarding the damage phenomena, mechanism, etcetera. But to verify preliminarily which our newly developed alloy shows the high avoid swelling resistance, we only use the iron irradiation in this study. But in the future, our group has a plan to put some specimen in the commercial nuclear reactor to connect with the real damage, the phenomena in microstructure aspect, and so between our preexisting TEM or the microstructure result.

Berta Oates
Thank you. Are there any more questions for Ji Ho?

What is the peak swelling temperature for stainless steel?

Ji Ho Shin
The peak swelling temperature is depending on lots of factors such as dose rates and the type of ion and material, etcetera. The definition of the peak swelling temperature is which temperature shows a high void swelling microstructure feature.

Depending on the temperature, at low temperature, only small defects are formed inside the matrix, such as black dots or the second [ph] for tetrahedral, but increasing the temperature, the microstructure evolution is higher thermodynamically. Some of the defects such as the voids and the interstitials meet together and form some cluster, which eventually forms voids. More the higher temperature, the individual voids are gathering and got some big void size. Depending on the temperature, the void size is different.

Meanwhile, depending on the temperature, as the size is increased, the density is dramatically decreased. The middle of the temperature is usually defined as the peak temperature in austenite stainless steel, and others such as nickel, pure nickel, steel, pure nickel alloy and pure Fe alloy. It's generally defined by the middle of the temperature. In stainless steel, around 10^{-3} or 4 dpa shows 500 to the 600 Celsius generally.

Berta Oates
Thank you. Again, congratulations on being the winner of the voted Most Popular Gen IV presenter for your research. Thanks again for sharing your expertise with us today.

How can you make a decision for true material for use in an alloy?

Ji Ho Shin
Could you speak again, please?

Berta Oates
How can you make a decision for true material for use in an alloy?

Ji Ho Shin
To approach the commercially used material such as the big companies at the large scale, it will be conducted large kind of the property test, such as SCC or such as the corrosion and mechanically test. Reproducibility [ph] is one of the issues which formed nanometer size precipitates is well maintained at the large-scale industry. The two tracks will be handled in the future to make a true alloy using the reactor in-core material.

Berta Oates
Thank you. Any more questions for Dr. Shin?

As a reminder, the presentation was recorded and the upload will be just a few days as it takes a few days to do some slight editing and rendering, and we'll get that presentation uploaded along with the slide deck, the PDF of the slides. Those are also available to you right now in the Handouts pane.

We have quite a list, a growing list of webinar presentations from the Gen IV. You have a handout in your Handouts pane showing the previous and a snapshot of the upcoming webinar presentations.

I don't see any further questions. Again, thank you Dr. Shin. Thank you for everyone who has joined and participated. We appreciate your interest in our topics. Again, if you'll give feedback on the survey that will follow, that does help us.

Patricia, do you have any closing thoughts or remarks?

Patricia Paviet
Yes, sure, of course. Thank you, again, Ji Ho, for your excellent presentation, and I would like to echo some of the people who have put the appreciation in the chat box about your presentation. It's always good for us that are aging to see junior workforce like you joining the nuclear energy industry. Like that we know that the future of this industry is in good hands. I personally would like to wish you the best for your postdoctoral research.

Again, thank you, everyone. Please join us next month, 15th of June, with Nuclear Waste Management Strategy for Molten Salt Reactor Systems.

Again, thank you, everyone. Wishing you a good day, or a good night.

Berta Oates
Thank you. Bye-bye.

Patricia Paviet
Goodbye.

Ji Ho Shin
Thank you.

END