Development of In-Service Inspection Rules for Sodium-Cooled Fast Reactors Using the System Based Code Concept
Dr. Shigeru Takaya, JAEA, Japan

Berta Oates
Welcome, everyone, to the next Gen IV International Forum Webinar Presentation. Today's presentation on the Development of In-Service Inspect Rules for Sodium-cooled Fast Reactors Using the System Based Code Concept, will be presented by Dr. Takaya.

Doing today's introduction is Dr. Patricia Paviet. Patricia is the Group Leader of the Radiological Materials Group Office at the Pacific Northwest National Laboratory. She is also the National Technical Director of the Molten Salt Reactor Program for the Department of Energy. She is the Chair of the Gen IV International Forum, Education and Training Working Group. Patricia?

Patricia Paviet
Thank you very much, Berta, for the introduction. Good morning. Good evening, everyone. It's a pleasure to have Dr. Shigeru Takaya with us. He is a Principal Researcher of Fast Reactor Cycle System R&D Center at the Japan Atomic Energy Agency. In 2003, he received his Doctor of Engineering degree from the University of Tokyo.

His main areas of research interests are structural integrity evaluation at elevated temperatures, maintenance technologies for SFRs, optimization of design and in-service inspection requirements on SFRs through plant life cycle based on the System Based Code concept.

He also works on the development of codes and standards for these fields and participates in several committees of ASME as well as JSME. He serves as the Chair of Subgroup on Elevated Temperature Design in JSME, and also, he is the Chair of ASME/JSME Joint Working Group on Reliability and Integrity Management Processes and System Based Code in ASME.

Without any delay, I thank you very much, Dr. Takaya, for volunteering and presenting this webinar. I give you the floor. Thank you so much.

Shigeru Takaya
Thank you, Patricia, for introducing me. Thank you for giving me this great opportunity. Good morning and good evening, everyone. Thank you for joining this webinar. I am Shigeru Takaya. I'd like to

talk about development of ISI rules for sodium-cooled fast reactors by using the system-based code concept.

This slide shows the contents of today's presentation.

At first, I'd like to explain the background. Then, ASME Boiler and Pressure Vessel Code Case N-875 will be explained. This code case was developed based on the system-based code concept and provides ISI rules for sodium-cooled fast reactors. After that, current status of development of ISI standards for SFRs, especially in JSME and ASME. Also, future visions will be explained. Finally, I'd like to summarize today's presentation.

Let's start with the background.

In the background, at first, I'd like to explain the importance of in-service inspection rules. After that, features of sodium-cooled fast reactors, SFRs, to be considered will be reviewed briefly. Then, conventional standard for ISI of SFRs will be explained. There were several problems with this standard, so new ISI rules for SFRs was developed based on the system-based code concept.

At the last of the background, I'd like to explain the essence of system-based code concept.

In-service inspection rules provide requirements for periodic inspections of passive components of nuclear power plants such as bases, piping, valves, and so on during the service, which is important for safety and stable operation. Effective and efficient ISI is crucial to suppress operation costs, which is one of the major components of power generation cost. ISI is important not only after the start of plant operation, but also even during the design stage.

Actually, ISI rules affect design of nuclear power plants because the accessibility to the components where ISI is required needs to be considered appropriately in the design. Sometimes, it might be more reasonable to change the design so that ISI can be exempted. Like this, ISI is important from various points of view. ISI rules need to be developed rationally by considering relevant features of reactor type and design of an individual nuclear power plant.

As you may already know, but sodium-cooled fast reactors have several different features from the conventional light water reactors. This table shows a comparison of operating conditions and the dimensions of reactor vessel as an example of the main components between PWR-type LWR and SFR, more precisely most prototype fast reactors in Japan. Of course, sodium is used as coolant material in

SFR. Sodium is opaque and chemically active, which makes it difficult to apply the traditional or conventional ISI technique to SFRs. Needless to say, visual tests using camera cannot be used on sodium, and conventional ultrasonic in-service examination is not as easily performed as in LWRs because of the limitation of accessibility.

On the other hand, sodium shows excellent compatibility with structure materials. In SFR, the corrosion is negligible impurity-controlled sodium, while the corrosion such as stress corrosion cracking is one of the main concerns in LWR. Reactor outlet temperature is above 500 degrees C. It's in creep regime. Creep and creep fatigue interaction damage should be paid attention as a typical degradation mechanism for SFR. High thermal stress and low pressure, and these features of the main zones are also related to failure modes to be considered in SFR. Like this, there are significant differences between LWR and SFR, so it is not reasonable to apply the ISI rules of conventional light water reactors to SFRs directly.

Historically, the American Society of Mechanical Engineers, ASME, Boiler and Pressure Vessel Code, Section XI, Division 3 had provided ISI rules for liquid-metal cooled plants, as shown here. Section XI, Division 3 was developed as part of the Clinch River Breeder Reactor Plant Project in the U.S. Clinch River Breeder Reactor was sodium-cooled fast reactor. It is considered that this Division 3 is for sodium-cooled fast reactor. However, the code revision was suspended due to the cancellation of the project. Several parts, including acceptance standards for examinations of Class 1 components were left as being in the course of preparation. That's practically difficult to apply to SFR plants.

To address this situation, ASME/JSME Joint Task Group for System Based Code was established about 10 years ago in the ASME Boiler and Pressure Vessel Code Committee. As a result of collaboration between experts from ASME and JSME, Code Case N-875 provides an alternative ISI requirement for liquid-metal cooled plants to Section XI, Division 3, was developed based on the system-based code concept and was issued in 2017. A fitness-for-service code for sodium-cooled fast reactor was concurrently being developed in JSME, and the first edition was approved last year.

Before explaining the details of Code Case N-875, I'd like to explain the essence of system-based code concept. By the way, the code or the system-based code concept does not mean the kind of simulation code, but kind of codes and standards, a set of provisions. The system-based code concept was originally proposed by Professor Asada about 20 years ago. There are various technical areas related to the structural integrity of components, such as load, material, design, inspection, and so on. The margin is given in individual technical areas kind of independently, and then accumulated like this. But how much is not clear, maybe too excessive.

On the other hand, in the system-based code concept, target reliability for the total integrity, structural integrity of the component is determined first. Then, the component is designed to meet the required reliability. The margins are distributed to each technical area. How to distribute the margin is flexible as long as total integrity meets the reliability. Margin exchange between different technical areas is also possible. Such margin exchange makes it easier to expand technical options.

If ordinary [ph] users want to employ the technical option with less margin, they only have to add additional margin in the other technical area to meet the target reliability. Such a flexible and consistent concept of the system-based code is very useful to develop the ISI rules suitable to the SFRs by considering the features of the SFRs.

From now on, I'd like to explain the ASME Boiler and Pressure Vessel Code Case N-875. At first, overview of the code case N-875 will be presented, and what ISI is required alternatively to Section XI, Division 3 will be explained. Then, logic flow to evaluate the applicability, and the key technical elements of the evaluation according to the logic flow will be explained. At the last part of this part, example will be presented.

As already mentioned, this code case was approved about 5 years ago. This code case provides alternative ISI requirement to Section XI, Division Three, using the system-based code concept. The code case is composed of the main body, provides automatic provisions of the logic flow to establish criterion for the application and two mandatory appendices. Appendix I is derivation of component target reliability from plant safety requirements. Appendix II is procedure for structural durability evaluation for passive components of liquid-metal reactors. Similar key parts of the code case have been incorporated in the new Section XI, Division, "Requirements for RIM programs for Nuclear Power Plants", as explained later.

This table shows a comparison of ISI requirements between Section XI, Division 3, and Code Case N-875. Code Case N-875 covers these five examination categories. Liquid-metal-retaining welds in Class 1 vessels protected or not protected by guard vessels. Liquid-metal-retaining welds in Class 1 piping protected or not protected by guard pipe or tank and internal components.

As for liquid-metal-retaining welds in Class 1 vessel or piping, continuous monitoring and VTM-2, a kind of visual tests are required by Section XI, Division 3, while just only continuous monitoring is required by code case N-875 if the conditions identified by using the SBC concept is satisfied. Acceptance standards for the continuous monitoring was in the course of preparation in Section XI, Division 3, but in the Code Case N-875, the acceptance standards for continuous monitoring were newly prepared.

As for internal components, VTM-3, another kind of visual test is required by Section XI, Division 3, while Code Case N-875 requires no ISI for internal components if the specific condition is satisfied.

The acceptance standard for continuous monitoring in the code case was prepared by using the related provisions in Division 3, and experience in Japan. Once leakage is indicated, it is required to conduct the confirmation of leakage in accordance with the procedure predetermined by the owner. If the confirmation takes longer time than the determined time, it is conservatively evaluated that the leakage is confirmed. In the case that leakage is confirmed, immediate shutdown of the system is required. In case the leakage is unconfirmed, it is considered the indication was due to fault of the monitoring system, so repair of the leak detectors is required to meet the minimum percentage of required working leak detectors.

This is the logic flow to establish criteria for application of the alternative ISI requirements, as explained maybe two slides before, are based on the SBC concept. The logic flow consists of two evaluations compensating each other. Stage I: Evaluation is structural reliability evaluation. Stage II is safety-related evaluation. Both probabilistic and deterministic approaches are applicable.

This logic flow is originally developed for this code case, but it could be for general use. Actually, it has been incorporated in the appendix for alternate requirements for NDE and monitoring in the new Section XI, Division 2.

I'd like to explain Stage I and Stage II evaluations in a little more detail.

In Stage I evaluation, the component level structure reliability and the design basis conditions are considered. Potential failure modes are determined based on the degradation mechanism, identified based on the inputs from these: Load, resistance, environment, and configuration. Component Level Requirements, CLRs, are determined in deterministic or probabilistic manner based on the input related to safety evaluation.

Component level requirements established in a probabilistic manner is called target reliability, in this code case. The reliability of the component is evaluated. At this moment, the contribution of ISI is not taken into account. Evaluated reliability and targets are compared. If the evaluated reliability meets the target, the CLR, the user may proceed to Stage II evaluation.

In the Stage II evaluation, the ability to detect flaws that ensures the plant can be safely shut down before the flaw reaches the maximum acceptable size is considered. Direct or indirect detection is allowable, which makes it possible to select suitable ISI technology flexibly according to the plant features.

In the case, postulated flaws are not detectable, and now we proceed to the reliability evaluation with a penalty. Its additional margin is demonstrated by imposing the penalty in the structural reliability evaluation, examination for flaw detection is not required, which is the margin exchange between different technical areas via the penalty.

From now, on, I'd like to explain key technical elements needed for the evaluation according to the logic flow. The first one is determination of failure modes. Degradation mechanisms that can potentially produce flaws during service is evaluated based on the list of the potential degradation mechanisms provided in the code case, as shown here, as well as operating and research experience. Failure modes are determined based on the identified degradation mechanisms. Failure modes not addressed in the design code are also considered, if necessary.

Next is determination of CLRs. Component Level Requirements, CLRs, are established either deterministically or probabilistically based on the input related to safety evaluation. Deterministically established CLRs are quantities such as the break size postulated in an accident scenario that define the allowable limits from the safety point of view. Probabilistically established CLRs are component level target reliabilities derived from quantitative plant level requirements available in quantities such as core damage frequency, containment failure frequencies, or large early release frequency. A method for derivation of component level target reliability is provided by Appendix 1 of the code case.

In Appendix 1, to derivate component level target reliability, a Probabilistic Risk Assessment, PRA, is used. PRA is usually used to integrate the individual reliabilities into the risk index. In that case, structural reliability is one of the inputs for PRA analysis. Risk indices such as CDF, CFF is output. On the other hand, the developed method used PRA in a reverse way to derive component level structure reliability from plant level risk target. In this case, target structural reliability is output, while risk target is one of the inputs. Target's structure desirability is determined to meet the risk target by using PRA analysis models under provided conditions.

To compare with the target, reliability needs to be calculated. But calculated structural reliability could vary depending on pitch and how uncertainties are considered. To narrow such scatters of calculation, Appendix II uniquely provides the procedures for structural reliability evaluation for passive components. The procedures consist of the failure scenario setting, modeling, and failure probability calculation.

JSME developed the guidelines on reliability of fast reactor components, as shown here; and the Appendix in the code case was developed based on the guidelines.

The next is evaluation of detectability of flaws. As explained, the basis of Stage II evaluation is detectability of postulated flaws. Not only conventional direct detection such as ultrasonic examination, but also indirect detection is allowable. Two new indices for indirect detection were introduced in the code case. One is maximum acceptable leak, MAL, which is applied for coolant boundary items. MAL is a leak that would not lead to increase in the CDF or CFF/LERF that has been calculated in the safety evaluation of the plant.

The other is unintentional discontinuity, UID, which is applied for non-coolant boundary items. UID is a change in the plant parameters such as temperature and velocity of coolant, that indicates that flaws exist before they lead to an increase in the CDF, CFF or LERF that has been calculated in the safety evaluation of the plant.

Like that, by expanding technical options, a suitable ISI method could be selected according to the plant features.

The last one is reliability evaluation with penalty. If postulated flaws are not detectable, reliability evaluation with a penalty may be conducted. If the result meets the target reliability, the examination is exempt. To show that the component has enough margin even without ISI, unrealistically conservative yet logically imaginable penalty which is correlated to the highest consequence failure mode is determined from the following four categories.

The first one is load. An additional load caused by failure of an adjacent component or part of the component that reduces the loads on the critical portion of the component.

Second one is resistance. A decrease in the resistance caused by loss of strength-enhancement mechanisms or metallurgical stability.

The third one is environment. A presumption that the component is subject to the most harmful environment postulated at the component location.

The last one is configuration and initial flaws, such as anticipated flaws or distortions.

So far, I explained the logic flow and the related key technical elements. From now on, I'd like to show some examples according to logic flow.

According to logic flow, actually, several trial evaluations have been conducted for the components of the prototype SFR in Japan, Monju, to illustrate the developed logic flow. Upper core structure, core support structure, reactor guard vessel, and class 1 piping.

Today, I'd like to show the result of core support structure and class 1 piping as an example of the probabilistic approach case, and the deterministic approach case, respectively.

The first one is core support structure. Probabilistic approach was employed in this case. This table shows the basic information of the core support structure of the Monju. The function is to maintain core configuration, and the CSS was made of Type 304 stainless steel and using sodium. Maximum temperature is about 400 degrees C. It's high, but under the creep regime. The main load is cyclic load by reactor start-ups and shutdowns.

CSS is categorized as internal components. Section XI, Division 3 requires VTM-3, which is visual examination intending to determine the general mechanical and structural conditions of components and their supports, and to detect discontinuities and the imperfections, is required; while Code Case N-875 requires no ISI if the condition is satisfied.

It is mentioned in the Section XI, Division 3, visual for example, periscope and light, or combination of the under-sodium scanning and dimensional gauging is available. But of course, conventional visual technique using periscope and light cannot be applied if the vessel is filled with sodium.

Firstly, failure modes have to be determined. Potential failure modes have to be analyzed exhaustively even if they are not addressed explicitly in the design code. First of all, corrosion in the purity-controlled sodium is negligible. The temperature is low enough to neglect creep damage. On the other hand, material properties might change due to the neutron irradiation. For example, degrees in ductility are one of the concerns, but a surveillance program during the operation is available to confirm the neutron irradiation effects.

The looseness of the fixing bolts may be another concern, but preventive measures against the rotation of bolts were taken in Monju. As a result, just fatigue damage due to cyclic loads by reactor startup and shutdown is left.

Next is the establishment of the CLR. Probabilistic approach was employed for CSS. Plant level requirements are needed. But those values are not available for Monju. However, such a discussion was made for demonstrations SFR in Japan, JSFR. Those numbers are used in this example. That is CDF is less than 10^{-6}, CFF is less than 10^{-7}. Then, plant level requirements are distributed by considering the plant PRA scope whether initiating events is internal events or external events, whether initial plant operating stage is during plant operation or shutdown state, and so on. These distribution factors are determined by the owner consistently with the plant safety design concept.

As for our example, initiating events is internal events. Initial plant operation stage is power operation. Failure of CSS directly results in core damage. The failure of CSS due to fatigue damage are categorized here. By using these numbers, and also considering design life of Monju is 30 years, as a result less than 6×10^{-6} was obtained as a target for failure of CSS due to fatigue.

In the stage I evaluation, crack initiation evaluation was conducted. Crack initiation was assumed when fatigue damage D_f equals to 1, n is number of cycles, and N_f is fatigue life. Evaluation position is the mount arm from the RV, as shown here.

Precisely speaking, this part is not part of the CSS, but except for this position, there are multiple load transfer paths to maintain the core structure. This part was selected as an evaluation position in this example. Direct Monte-Carlo method was employed for calculation method. Number of samples is 10^9, and thermal stress and fatigue life was chosen as random variables.

The result is shown here. The horizontal axis is the fatigue damage, D_f, and here is the criterion for the crack initiation. The vertical axis is 1 minus cumulative probability. As shown here, the number of crack initiation samples was zero out of 10^9. It was shown that the component CSS has sufficient reliability, so we proceed to stage 2 evaluation.

In stage 2 evaluation, detectability of flaws is evaluated at first. However, the appendix for the under-sodium scanning in Section XI, Division 3, was in the course of preparation. It was assumed that the under-sodium scanning systems were not available yet in this evaluation. It was also assumed that there were no monitoring methods to detect flaws in CSS. We proceed to reliability evaluation with penalty.

In reliability evaluation with penalty, a fully circumferential crack with depth of 10% of the thickness, that is too large to be missed, was assumed as an initial defect.

The crack growth evaluation was conducted. 50% of wall thickness was selected simply as failure criterion. Direct Monte-Carlo simulation was conducted. Number of samples is 10^9, and thermal stress and co-efficient in this equation was selected as random variables.

The results are shown here. Horizontal axis is crack depth. Vertical axis is 1 minus cumulative probability. Initial depth is here, and the failure criterion here. As you can see, the crack hardly propagated. The number of failure sample was zero out of 10^9. It is shown that component has sufficient reliability even with the penalty. Based on this evaluation, it was concluded that alternate requirements of the Code Case N-875 is applicable to CSS Monju, that means no NDT is required.

Next one is second example. Second example is primary heat transfer system piping. The function is primary heat transfer, and this piping was also made of Type 304 stainless steel. Inside of the piping is sodium, and outside is inert gas. Maximum temperature is about 500 degrees C. It's in creep regime. The main load is cyclic load by reactor startups and shutdowns. This piping is categorized as a liquid-metal-retaining Class 1 piping protected or not protected by guard pipe or bending of the position.

Section XI, Division 3, requires continuous monitoring, and VTM-2, which is the visual examination intending to detect accumulations of liquids, liquid streams, liquid drops, and smoke. On the other hand, Code Case N-875 requires only continuous monitoring if the SBC condition is satisfied.

Based on the similar discussion as the previous example, a crack initiation and propagation due to fatigue-creep interaction damage was selected as a failure mode for PHTS piping.

Next, deterministic CLR is established. The break size postulated in the plant safety evaluation can be used as a deterministic CLR. As for the PHTS of Monju, a wall-through crack with an opening area of 22 square centimeters was postulated in the plant safety evaluation. This wall-through crack with the opening area of 22 square centimeters was selected as a deterministic CLR for PHTS piping.

In the previous example, the probability of crack initiation was evaluated. But as a matter of fact, fatigue as well as fatigue-creep interaction damage is already addressed by the design code. Therefore, fatigue-creep interaction damage of PHTS piping is restricted below the design allowable level. It is considered that the crack would not initiate and propagate to the size determined as CLR, in the conditions of the PHTS. Such a deterministic approach was applied. We can proceed to stage II evaluation.

In stage II evaluation, detectability of flaws is evaluated. In Monju, two sodium leak detection systems are installed for small-scale leaks. One is Sodium Ionization Detector, SID, which detects a leak by monitoring the change of ion current produced by ionizing sodium aerosol in an inert gas atmosphere. The other one is Differential Pressure Detector, DPD, that detects a leak by monitoring a change of difference pressure at the filter installed in the detector. These detectors are designed to detect leaks of 100 grams per hour.

Next, MAL, Maximum Acceptable Leak should be determined. In the plant safety evaluation of Monju, a through-wall crack with an opening area of 22 square centimeters, this was selected as CFR, was postulated. The leakage rate just after the accident was evaluated to be at approximately 80 kilograms per second. As you can see, the capability of leak detection of SID and DPD is much greater than MAL, determined MAL.

In addition to the capability of leak detectors, demonstration of Leak-Before-Break is essential to show the effectiveness of the continuous monitoring. Leak-before-break is a concept that is leakage is detectable before a break, so plant can be shut down safely. In this example, guidelines for LBB assessment of SFR provided by JSME was used. This is for procedures provided in the guidelines. Unstable fracture assessment, leak rate assessment, and penetrated crack assessment was conducted. If critical crack range, CR, is larger than both detectable crack length, C_d; and penetrated crack length, C_p; it is considered LBB is demonstrated.

As for this example also, LBB is successfully demonstrated. It was shown that postulated flaws are detectable indirectly. Based on this discussion evaluation, it was concluded that alternative requirements of Code Case N-875 is applicable. That means just continuous monitoring is required for the PHTS piping.

Here, I would like to explain current status of development of ISI standards for SFRs in JSME and ASME briefly.

In JSME, standards related to ISI of SFRs are developed based on the SPC concept. As already mentioned, fitness-for-service code for sodium-cooled fast reactors was approved last year. General rules and ISI requirements of Class 1 components and their supports are provided in the First Edition. Continuous monitoring is required to coolant boundaries as Code Case N-875.

At the same time, guidelines for LBB assessment of sodium-cooled fast reactors are developed and approved. The guidelines are available to determine the required sensitivity of leakage detectors for continuous monitoring. The guidelines for reliability evaluation of sodium-cooled fast reactor components are also developed. These are unique guidelines providing procedures of reliability evaluation which is important for the system-based code concept.

In ASME, brand-new fitness-for-service code for all types of nuclear power plants, Section XI, Division 2, Reliability and Integrity Management was published in 2019. Section XI, Division 2 provides technology-neutral requirements and supplements for specific types of nuclear reactors – currently for light water reactor and high-temperature gas-cooled reactor.

This is a flowchart of the RIM. But as you can see, RIM shares the basic concept with system-based code concept. Actually, some parts of the code cases have been already incorporated in Division 2. The procedures of deriving target reliability in the code case are incorporated as a mandatory Appendix II in Division 2. The logic flow and reliability evaluation procedures are included in the non-mandatory Appendix A.

As I said, currently, supplements for the sodium-cooled fast reactors were not included in Division 2, but development of supplement for liquid-metal sodium cooled fast reactors is one of the top priority action items of Section XI committee. Therefore, ASME/JSME Joint Working Group on RIM Processes and SBC is now developing the supplement based on the ASME Section XI, Division 3, Code Case N-

875, and JSME Standards, including LBB Assessment Guidelines for SFRs.

Lastly, I'd like to talk about future vision.

This figure is the image of the structure or hierarchy of the standards. At the top of that is safety goals. The safety standards are prepared to achieve the safety goals. Structural standards are prepared to support safety standards. However, the relationship between structural standards and the safety standards sometimes is unclear. Sometimes, it is difficult to say what safety function is implemented by following certain provisions in structural standards.

A closer link between safety standards and structural standards is essential to develop nuclear power plants that balance safety and economic efficiency at higher level. Reliability target or target reliability is a promising key concept to link both standards. Also, technology-inclusively development of standards is important. It's essential to develop standards for various types of advanced reactors efficiently in a timely manner.

The structure of RIM, the combination of common part is technology-neutral part and the specific part. Specific rules for each reactor part are very reasonable. Such a structure is important from the view of explainability. If the common parts are shared among all types of reactors, the adequacies of the common part cannot be explained repeatedly. As explained in this presentation, the flexible concept of SBC is expected to work as a basic principle to develop technology-inclusive standards, and also specific rules for each reactor.

Summary: Effective and efficient ISI is crucial for safety, stable operation, and economic efficiency of nuclear power plants. SFRs have several desirable features such as excellent compatibility between sodium and structural materials while traditional volumetric and surface tests are not as easily performed as in light water reactors. The system-based code concept can be used to develop ISI rules rationally by considering relevant features of reactor type and design of individual nuclear power plant. ASME Boiler and Pressure Vessel Code Case N-875 is a good and important example of development of Codes and Standards based on the SBC concept. The details are introduced in this presentation.

Recently, the new Section XI, Division 2, RIM has been developed for various advanced nuclear reactors. RIM shares key concepts with SBC, and the case has been partially incorporated in RIM. JSME has also developed a new fitness-for-service code and related guidelines based on that SBC concept.

Link between safety standards and structural standards, as well as development of technology-inclusive standards are important for advanced reactors. SBC concept is expected to work as basic principles.

This is the end of my presentation. Thank you for your attention. I am happy to answer questions. Thank you.

Berta Oates
Thank you. Thank you, Dr. Takaya. As questions are coming in, we'll take a quick look at the upcoming webinar presentations.

In October, a presentation on the Sodium Integral Effect Test Loop for Safety Simulation and Assessment, abbreviated as STELLA. Again, you have the registration link and the information flyer describing that webinar in the handouts pane today.

In November, a presentation on the Visualization Tool for Comparing Energy Options.

In December, The Mechanisms Engineering Test Loop Facility at the Argonne National Laboratory.

We do have some questions that have come in. Dr. Takaya, I have validated you so that in the questions pane, you can undock that. Sometimes, it's easier to see the questions, as I read them aloud. The first one is, can this methodology or code be used to other liquid-metal cooled reactors like liquid lead or LBE?

Shigeru Takaya
Thank you for the question. As I mentioned, the strategic or the logic flow is applicable to other types of reactors, not only liquid-metal or other types of reactors. Actually, the logic flow was included in Section XI, Division 2, which is not limited to that target reactor. But of course, specific methods, such as crack growth evaluation or something like that, are not directly applied. The essence or the key part of the methodology can be used.

Berta Oates
Great! Thank you. The next question – do you expect that the owners or designers will build additional margins, for example, perhaps, a bolt backing out of the core plate is not a safety issue, but it could be very expensive or time consuming to repair. The related question is – do you anticipate that the regulator will want to review the additional margin elements as some part of the defense in depth?

Shigeru Takaya
Thank you. Do you expect that the owners or designers will build in additional margins, perhaps bolt? Okay. Do you anticipate that the regulator will want to review the additional margin elements as some part of the defense in depth? It's a good question, and an important question. This code case is focused on safety issues, but in the same manner, such kind of issue can be considered. For example, the stable operation, not safety issue, but some time-consuming manner or something like that.

But I think what is important is that what is the requirement for safety issues and what is the requirement from the other objectives? How to determine the target is related to these issues. The logic flow or the methodology to derive the target reliability is kind of answer to that question.

Do you anticipate that the regulator will want to review the additional margin elements as some part of the defense in depth? Defense in depth is also important point. As I said, the logic flow consists of two parts: Stage 1 evaluation and stage 2 evaluation. Even if stage 1 is cleared, that means evaluated reliabilities, means the target reliability, it is required to proceed to stage 2 evaluation from the safety-related point of view. In such manner, the defense in depth is also considering developing this logic flow.

Berta Oates
Thank you. What is the definition of Class 1 components?

Shigeru Takaya
Class 1 components are the reactant coolant boundary.

Berta Oates
Thank you. If you have additional questions, go ahead and type those in, and while we are waiting, I'll wait just another minute. Again, thank you, Dr. Takaya, for your time and sharing your expertise. It's been a pleasure to get to know you during this webinar presentation. We look forward to hearing more from your efforts in the future.

Shigeru Takaya
Thank you very much.

Berta Oates
I don't see any additional questions coming in. There is a link for the email if you have something that comes to mind later. If you want to reach out directly, you can do that. Since there are further

questions, I think we'll go ahead and end it for today. Patricia, do you have any closing thoughts?

Patricia Paviet
No. I would love to just say thank you to Dr. Takaya, again, for volunteering to give this webinar. I would like to thank the audience to participate in all these webinars. We'll meet you again. As Berta said earlier today, you have the possibility of a survey to give us some thoughts about how we are doing and also suggestions of webinar that you would like to hear about or understand a little bit better about the subject.

Thank you very much, everyone. Thank you again, Berta. Thank you, Dr. Takaya. Wish you a very good day, very good night. Bye-bye, everyone.

Berta Oates
Bye-bye.

Shigeru Takaya
Bye-bye. Thank you.

END