In-core-instrumentation methods for 3-Dimensional distribution information of reactor core temperatures and Melt-down

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Abstract: The tsunami-induced nuclear accident at Japanese Fukushima power plants in March 2011 has revealed some weaknesses in the severe accident monitoring system. The plant instrumentation did not provide utility, safety experts, and government officials with adequate and reliable information. The information on the reactor core damage and coolability is critical for making decisions correctly as well as in a timely manner during the course of the mitigation of severe accidents. Current Pressurized Water Reactor (PWR)s have an In-Core-Instrumentation (ICI) system that measures the temperature distribution of the top surface (*i.e.* Core Exit Temperatures) of the reactor core mainly to indicate when to begin Severe Accident Mitigation Guidelines (SAMG). This design concept giving only the core exit temperature has limitations in terms of sufficiency as well as availability of the information necessary for diagnosis on the status of the degraded core and the effectiveness of the measures taken as mitigation strategies. The reactor core exit temperatures are not sufficient to support the assessment of the degree of the core damage and the location of the molten core debris and recognition whether the core damage progresses on or it is mitigated. The ICI location being at the top of the reactor core also makes the ICI thermocouples vulnerable to melt-down because the upper part of the reactor core uncovers first, thereby melt down at the early stage of the accident. This means that direct indication of reactor core temperature will be lost and unavailable during the later stages of severe accident. To address the aforementioned weaknesses of the current ICIs, it is necessary to develop a new ICI system that provides information that is more expanded and more reliable for accident mitigation. With the enhanced ICI information available, the SAMG can be prepared in more refined and effective way based on the direct and suitable indication of status of damages and the 3-dimensional temperature profile of the core rather than guesses and assumptions. Furthermore, this goal needs to be achieved economically and with minimal changes to current design of reactor and its instrumentation that has been proven and well established through many years of operation. In this paper, methods for a new ICI system to provide three-dimensional view of the reactor core temperatures and melt-down are introduced.

Keyword: in-core-instrumentation; core exit temperature; severe accident, core cooling

1 Introduction

The tsunami-induced nuclear accident at Japanese Fukushima power plants in March 2011 has revealed some weaknesses in the severe accident monitoring system. The plant instrumentation did not provide utility, safety experts, and government officials with adequate and reliable information. The information on the reactor core damage and coolability is critical for making decisions correctly as well as in a timely manner during the course of preventing and mitigating severe accidents. During the Fukushima accident, it was not possible to know the status of reactor core. Many experts suggested different and split opinions on how much

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of the reactor core had been melted. Some experts even suggested that steam explosions due to destruction of the reactor are likely to occur. The fact that it was not possible to assess, with certainty, the status of reactor core was one of the key factors that contributed to the mitigation failure of the 2011 accidents at Fukushima nuclear power plants. Considering the experiences at the Fukushima NPP as mentioned above, reactor In-Core Instrumentation (ICI) that is important for the prevention or mitigation of severe accidents were evaluated if its upgrade is necessary for the development of a meaningful severe accident management program. The upgrade will be meaningful if a severe accident management program that reduces risks in an appreciable way can be developed. In this paper, a new ICI upgrade approach to monitoring reactor core during severe accident is suggested to better support operators and decision makers in preventing and mitigating severe accidents with more reliable and more informative reactor in-core instrumentation.

2 Current ICI design and its issues

Current Pressurized Water Reactor (PWR)s have an In-Core-Instrumentation (ICI) system that measures the temperature distribution of the top surface *(i.e.* Core Exit Temperatures) of the reactor core mainly to indicate when to begin Severe Accident Mitigation Guidelines (SAMG). [1-4] The current In-Core Instrument assembly consists of five rhodium emitter detectors, one K-type thermocouple made with 2 cables, a background detector and eight filler cables wrapped helically around an inner central member and encased by an outer sheath tube as shown in Fig. 1.

The cross section of the current ICI is as shown in Fig. 2. The outer sheath tube separates and protects the detectors and thermocouple from the reactor coolant. The outer sheath tube is connected to the header/seal plug through an adaptor, which provides the sealing surface to the reactor pressure boundary. The detectors, thermocouple and central member are routed out of the reactor core to their respective termination points through the header. Exiting at the top end of the seal plug, the thermocouple and detectors are enclosed in a flexible metal hose at the end of which an electrical receptacle is connected. 8 signal wires go through the header that can be vulnerable to the reactor pressure boundary. For this reason, the header and seal plug are tested for its sealing characteristics during manufacturing.

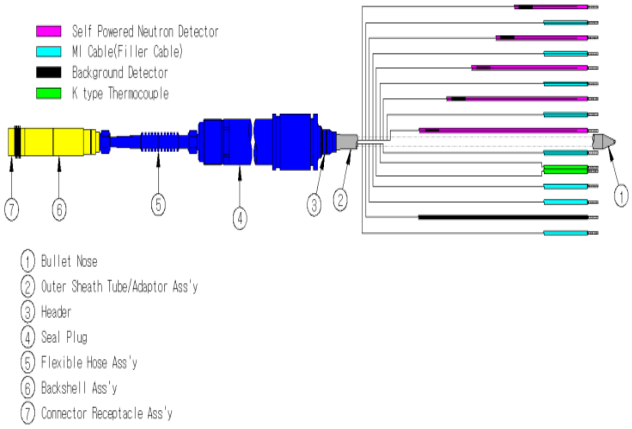

Fig.1 Arrangement of current ICI.

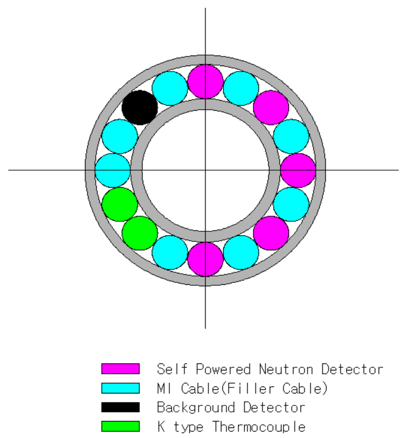

Fig.2 Cross section of current ICI.

ICI Assembly is installed from the Seal Table to the reactor core through the Guide Tube as shown in Fig. 3.

This design concept giving only the core exit temperature has limitations in terms of sufficiency as well as availability of the information necessary for diagnosis on the status of the degraded core and the effectiveness of the measures taken as mitigation strategies. The reactor core exit temperatures are not sufficient to support the assessment of the degree of the core damage and the location of the molten core debris and recognition whether the core damage progresses on or it is mitigated. The ICI location being at the top of the reactor core also makes the ICI thermocouples vulnerable to melt-down because the upper part of the reactor core uncovers first, thereby melt down at the early stage of the accident. This means that direct indication of reactor core temperature will be lost and unavailable at the later stages of severe accident.

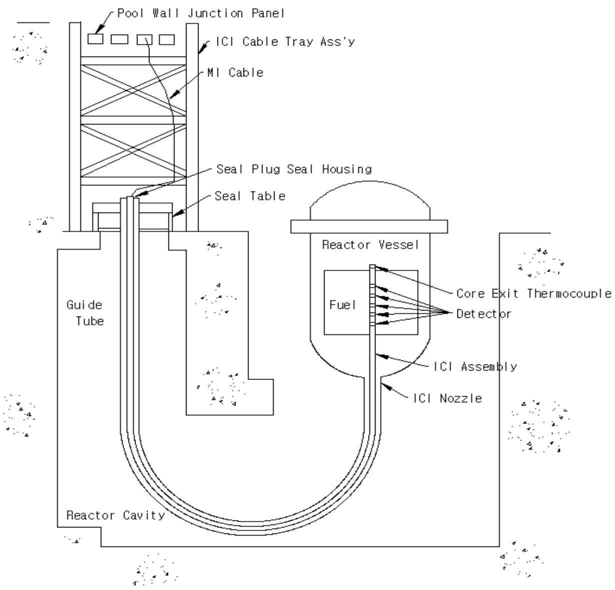

Fig.3 Installation of ICI system.

3 Parameters for evaluation of severe accident states from 3D ICI

To address the aforementioned weaknesses of the current ICIs, it is necessary to develop a new ICI system that provides information that is more expanded and more reliable for accident mitigation. With the enhanced ICI information available, the SAMG can be prepared in more refined and effective way based on the direct and suitable indication of status of damages and the 3-dimensional temperature profile of the core rather than guesses and assumptions.

3.1 Test of K-type thermocouple behavior during severe accidents

During severe accidents reactor core temperature can eventually rise over 3000 ^{o}C when all the reactor fuels and internals will be melted to form a molten core. To investigate the possibility of using ICI thermocouples in this condition, a test has been performed to confirm the behavioral characteristics of the K-type thermocouples when it is exposed to high temperatures with molten core. A specimen of thermocouple (Fig. 4) was slowly inserted into the 1500℃ molten metal in the blast furnace (Fig. 5). This insertion of specimen thermocouple (TC) wire repeated three times for measurement and recording of temperature, voltage, and resistance, respectively. The non-simultaneous recordings are as shown in Fig. 6, 7 , and 8.

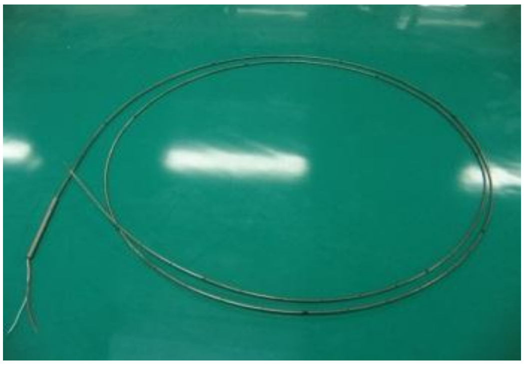

Fig.4 K-type thermocouple specimen.

Fig.5 High temperature blast furnace.

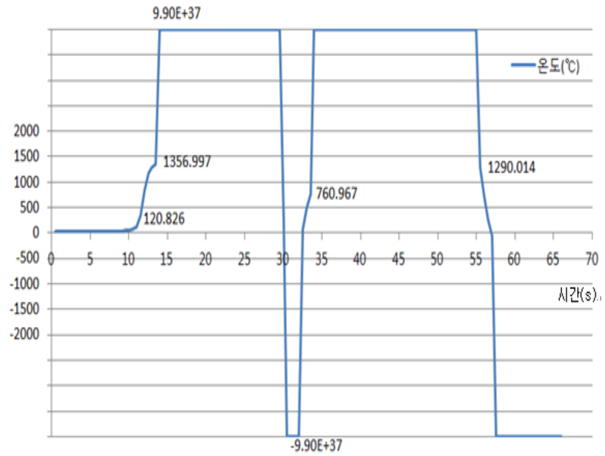

Fig.6 Temperature recording.

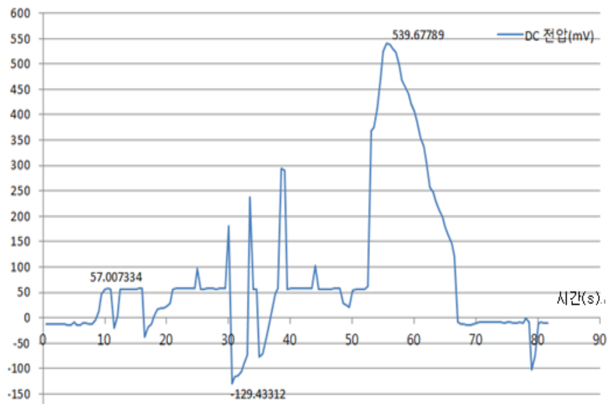

Fig.7 Voltage recording.

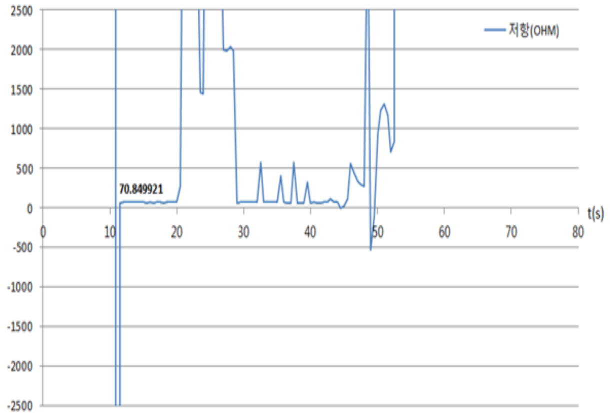

Fig.8 Resistance recording.

3.2 TC indication for reactor core states

From the data collected from the test, following phases of TC temperature/melt reading has been identified. Further test is scheduled next for stricter verification of TC reading changes as the reactor core goes through the heating, melting and cooling phases.

Phase 1: Normal Reading before Melted TC

- Condition: Thermocouple temperature less than 1260 ^{o}C and TC wire has not been melted
- TC reading: Normal reading

Phase 2: Inaccurate Reading before Melted TC

- Condition: Thermocouple temperature between 1260 ^{o}C and 1350 ^{o}C and TC wire has not been melted
- TC reading: Inaccurate but useful reading

Phase 3: Unstable Reading with Melted TC

- Condition: Thermocouple temperature over 1350 ^{o}C, TC wire has been melted, and TC contact alternates between open state and closed state
- TC reading: Very inaccurate and high reading when TC contact is closed and fluctuating positive/negative reading when TC contact is open.

Phase 4: TC contact stays open after TC Melted and Cooled

- Condition: TC wire has been melted, cooled and TC contact stays open
- TC reading: Fluctuating positive/negative reading

Phase 5: Inaccurate Reading with TC Melted, Cooled, and its contact stays closed

- Condition: Thermocouple temperature between 1260 ^{o}C and 1350 ^{o}C, TC wire has not been melted and cooled, and TC contact stays closed.
- TC reading: Inaccurate but useful reading

Phase 6: Normal Reading with TC Melted, Cooled, and its contact stays closed

- Condition: TC wire has been melted and cooled to temperature less than 1260 ^{o}C
- TC reading: Normal reading

4 Application of 3D information of ICI outputs for severe accident management

With the current approach based on parameters measured from outside of Reactor Vessel, there are limits in evaluating the severe accident progression states such as core cooling, oxidation, core damage and relocation, This is due to lack of models and knowledge, and excessive uncertainties involved. Hence, the direct sensing the inside of the Reactor Vessel using 3 Dimensional Temperature Distribution (3DTD) of reactor core can be very useful in assessing accident progression states and determining mitigation actions. 3DTD can be used as backup instrumentation for reactor core temperature and reactor vessel level in case primary instrumentation is not available for any reason.

4.1 Information for assessing core cooling

Inventory of coolant around the reactor core in reactor vessel can also be evaluated by checking if the temperatures of 3DTD along a horizontal line are around or beyond saturation temperature. 3DTD also provides information on how much and where the intact reactor core is located. This information will be used to determine whether feed/bleed operation will be effective.

4.2 Information for assessing core oxidation and radioactive material generation

3DTD can provide data to evaluate how much of the reactor core has been damaged (*i.e*. How much of the fuel assembly has been oxidized and melted). This degree of reactor core damage translates to the information on how much hydrogen and radioactive substances has been generated from the reactor core.

4.3 Information for assessing molten core relocation and threat for molten core ejection

3DTD provides data to evaluate the location of molten core. The location of molten core can be used to evaluate whether or when reactor vessel will fail. The location of molten core also tells if core cooling through feed and bleed will work or not. 3DTD data can be used to estimate how fast the core damage is progressing. This damage rate determines the time available for mitigation of accident and evacuation of public to safe area.

4.4 Application of 3DTD for prevention domain of SA

The prescriptive Emergency Operating Procedure (EOP) is performed to prevent incidents and accidents from degrading and escalating to severe accidents. 3DTD will provide very good feedback information on whether the feed/bleed operation in EOP is effective to limit the damage to the reactor core. Entry to the SAMG can be timelier because SA can be declared even if the CET temperature is lower than the threshold (650 ^{o}C) used today when the rate and degree of core damage is high. The declaration of SA can be delayed even if the CET temperature is higher than the threshold (650 ^{o}C) used today when the rate and degree of core damage is low.

4.5 Application of 3DTD for mitigation domain of SA

With 3DTD information, more suitable mitigation actions that fit the detailed situation in reactor core can be chosen with no or reduced possibility of negative effects of the mitigation actions. 3DTD information also rules out or reduces the possibility that there are too many opposing expert opinions and takes considerable effort and time to reconcile these differences before selecting mitigation actions.

4.5.1 Terminating the progress of core damage

The temperature distribution provided by 3DTD is a prompt and direct feedback to determine the effectiveness of the core cooling after coolant injection into the reactor core. The location and degree of intact core is crucial in determining if core cooling through internal injection is feasible. Location of molten core is also important in determining if core cooling through internal injection is feasible.

4.5.2 Maintaining the integrity of the containment

The degree and rate of fuel cladding oxidization evaluated from 3DTD allows an accurate estimation of hydrogen generated from the core. Hydrogen concentration estimation allows timelier actions to prevent hydrogen explosions.

The location of molten core, degree of intact core, and reactor vessel coolant inventory allow experts to estimate the time before reactor vessel failure Based on this time, External Reactor Vessel Cooling (ERVC) operation can be initiated at the right time.

4.5.3 Minimizing the releases of radioactive material

The rate of core damage will can be used to predict the concentration of hydrogen and radioactive substances. If the high level of hydrogen is expected, the necessity to release radioactive material to the public can be decided early in the progress. This will allow enough time to evacuate the public to the outside of the controlled area. This will also allow opening vent early enough based on hydrogen estimation that is higher than normal.

5 New ICI design

In this paper, methods for a new ICI system to provide three-dimensional view of the reactor core temperatures and melt-down are introduced. To lower the cost and impact to the current ICI design that has been proven, this new ICI design is developed by augmenting the single layer ICI design currently used in PWRs. In the existing ICI design filler cables serve no functional purpose and are used to fill the space only. So, the filler cables are substituted with thermocouples installed at different heights selected so that each thermocouple is associated with unique location in terms of height of the reactor core as shown in Fig. 9. The measuring junctions of the thermocouples are located in the middle of four guide tube dimples in the core and Lower Head Plenum so that the thermocouple can measure the temperature near the measuring junctions with fast response time. Measuring junctions are also located at ICI penetration of the Reactor Vessel Head so that it can indicate the threat for molten core ejection from Reactor Vessel. Cross section of the new ICI design is shown in Fig. 10.

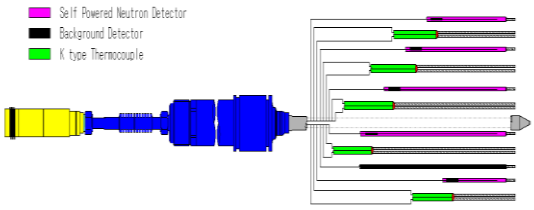

Fig.9 Arrangement of new ICI.

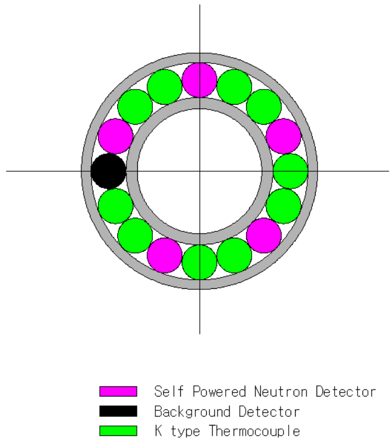

Fig.10 Cross section of new ICI.

6 Issues of new ICI design

The upgraded ICI design should be such as to ensure appropriate independence from existing systems with regard to the use of the instrumentation under accident and/or severe accident conditions. Independence with the added thermocouples was evaluated for the Self-Powered Neutron Detectors (SPND), background detector, and ICI components (*i.e.* header and seal plug) that provide the sealing surface to the reactor pressure boundary.

6.1 Electrical independence

Electromagnetic interference among the added TCs, SPNDs and background detector has been evaluated that the interference is minimal and does not significantly influence the performance of each system. The influence from TCs to SPNDs exists in current design with no significant impact to SPNDs. The fact that the electric currents in TC and SPND change very slowly and their magnitude is very small, allows us to conclude that their crosstalk is very small and insignificant. In the proposed ICI design, the SPNDs and TCs are arranged in such a way that each SPND is surrounded by two cables of TC whose current directions are reversed and hence, the electric currents induced at the SPND by the two cables of TC are canceled and approximate to zero.

Furthermore, any influence from the added TCs will be measured by the background detector and will be compensated appropriately.

6.2 Mechanical independence

The weight changes due to the substitution of MI cables with TC cables influences the seismic qualification of ICI assembly. Therefore, preliminary analysis based on a calculation method has been performed and it has been concluded that the impact of weight changes to seismic qualification is minimal and acceptable with current design. Impact to the sealing characteristics due to the added thermocouples is considerable because the header of ICI assembly, which is the crucial part of ICI for sealing, has 16 holes to accommodate 8 thermocouple wires from 8 holes in current design. The test described in Section 7 was done to address this sealing issue.

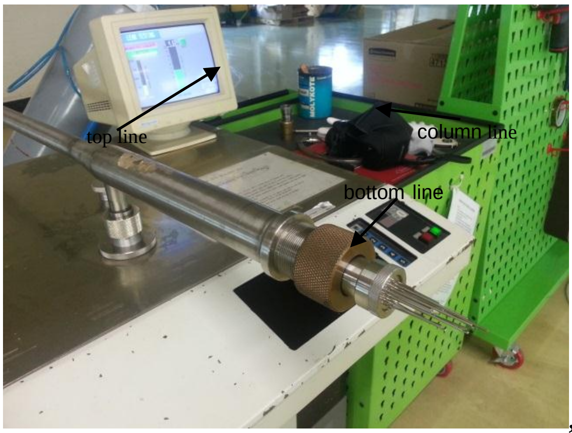

Fig.11 Helium leak test setup.

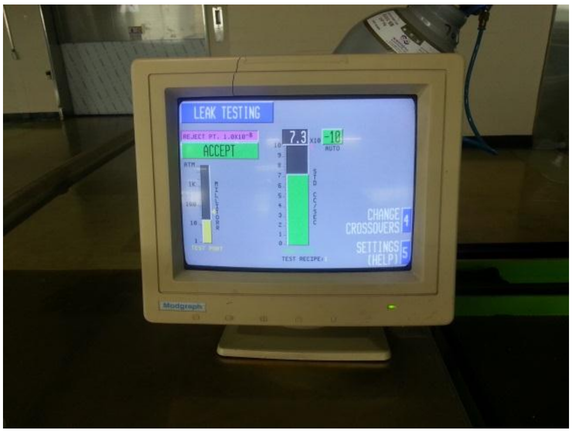

Fig.12 Seal performance test result.

7 Pressure integrity test of ICI header

ICI passes through the reactor pressure boundary and is a part of the in-vessel section that must be designed according to ASME codes. It consists of the seal plug, header, thermocouples and detectors, and all associated joints. We redesigned the header to accommodate the added thermocouples and to maintain the seal function at the pressure boundary. Header assembly with 5 thermocouples and detectors was manufactured and tested if it meets the qualification requirements in accordance with ASME Section Ⅸ. A helium leak test was performed as shown in Fig. 11. Leak rate results were within limits as shown in Fig. 12 and test was completed with success.

8 Conclusions

Three dimensional augmentation of existing ICI to support the prevention and mitigation of severe accident has been introduced in this paper. Its potential benefits in reducing the risks by minimizing uncertainties inside the reactor core and maximizing available time for mitigation in severe accident management has been discussed qualitatively. This approach will allow more sophisticated responses possible due to the direct and clear indication on the reactor core damage is provided in a prompt manner whereas very crude responses based on conservative assumptions will be made in a delayed manner when without 3DTD. The cost effectiveness of the approach is persuasive considering the fact that the proposed approach for ICI in this paper makes no changes to the reactor core including reactor core internals, ICI guide tubes, and ICI outer sheath tubes. The ICI filler cables, which do not serve any functional purposes, are replaced by additional thermocouples at desired heights of each ICI. The electrical and mechanical impact, including seal performance, of introducing added thermocouples in current ICI has been evaluated to be insignificant. Further research is necessary to evaluate the contribution of this technology to the reduction of risks in severe accident management, quantitatively; to evaluate the cost to supply the technology; to further clarify the behavior of the thermocouple in the evolution of reactor core in severe accident; to develop further guidelines and bases for using 3DTD for SA management. This new approach is applicable not only for new power plant but also for operating plants in Korea as well as in other countries and can potentially contribute to the safety of nuclear power plant substantially.

References

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