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Information Needs and Instrument Availability for Accident Management : Application to YGN 3&4

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Abstract

This paper introduces the five-step methodology for identifying information needs and assessing instrument availability during the course of severe accidents in nuclear power plants. The methodology is applied to the Yonggwang (YGN) 3&4 to shed light on accident management. It constructs three safety objective trees to prevent the reactor vessel failure, to prevent the containment failure, and to mitigate the fission product release from the containment. The study assesses information needs and instrument availability under severe conditions for preventing the reactor vessel failure of YGN 3&4, and recommends additional instruments that may prove to be of vital importance in managing the accident.

1. Introduction

The potential severe accident sequences and phenomena threatening the plant safety have been identified by applying the probabilistic safety assessment (PSA) techniques to nuclear power plants. Successful management of severe accidents requires the information about the plant status so that the plant personnel may diagnose the occurrence of an accident, monitor the status of the plant, select strategies to prevent and mitigate the safety challenge, and implement appropriate strategies. However, since existing safety-related instruments installed in nuclear power plants are primarily designed for managing design-basis accidents (DBAs), it is required to assess the availability and adequacy of the instruments during severe accident conditions.

This paper introduces the five-step methodology to identify nuclear power plant information needs and to assess the availability of the instruments during severe accidents [1]. This methodology was illustratively applied to severe accident management (AM) for YGN 3&4 nuclear power plants.

2. Methodology

This section presents a brief procedure to identify information needs and to assess the availability of instruments required to manage severe accidents. Figure 1 shows the task procedure to be presented here.

The first task is to examine the accident sequences and vulnerabilities of the nuclear power plant spanning initiating events, core damage, and severe accident phenomena causing failure of the reactor vessel and containment. This task, performed through individual plant examination (IPE) of the specific plant, develops AM strategies to terminate the accident progression, and to prevent the severe accidents and mitigate the consequence of the events.

The second task is to determine the information needs to diagnose the plant status, and to implement AM strategies and monitor the effect of the implemented strategy on the plant. There are two alternative approaches to achieve this task : one is to analyze the identified accident sequences and the other is to use the safety objective tree (SOT). The first method analyzes the accident sequences to determine the information required for the operators to diagnose the plant status sufficiently and implement AM actions effectively with detailed examination of the identified accident sequences. The second method uses SOT appearing as a tree with hierarchical structure. It starts from the overall safety objective of AM to prevent reactor vessel failure, to prevent containment failure, and to mitigate fission product (FP) release. Safety functions must be maintained within the safety limit to fulfil the safety objective. Challenges threatening safety functions, mechanisms related to physical phenomena or causes of challenges, and AM strategies to prevent or mitigate the consequence of the mechanisms are to be identified. After constructing SOTs related to each of the AM goals, we can determine all the information needed to monitor whether the safety functions are maintained within the boundary of the safety limit and whether the challenges threatening the plant safety are present. We can then select appropriate AM strategies and initiation time, implement the selected strategy, and observe whether the implemented strategy has a desired effect on the plant.

The third task is to examine instruments supplying the information required from the second task. Examination of instruments includes the identification of whether the instruments supplying the required information are being installed in the plant or not, and of the instrument capabilities such as measurement range and environmental qualification limit, and so on.

The fourth task is to analyze plant physical parameters under severe accident conditions in all local areas including the reactor coolant system (RCS), containment building, turbine building, auxiliary building, and so on. Those parameters to be considered are temperature, pressure, humidity, and radioactivity. It should be considered failures of the instrument support systems, such as failures of AC, DC and battery power supply, service water and instrument air supply, and so on.

The fifth task is to assess the instrument availability by comparing the parameters representing instrument capability identified in the third task with the plant process parameters in the fourth task. It is to determine whether the instrument capability falls within the measurement range and environmental qualification limit, and whether instrument supply systems fail or not. It then checks out which instruments will be available under severe accident conditions, and determines the time when the instruments may fail or their performance be degraded, and which instruments must be installed additionally to supply the required information.

3. Application

Assessment of information needs and instrument availability for severe accident management of YGN 3&4 is performed according to the task procedure presented in section 2.

3.1. Accident Sequences and AM Strategies

According to the IPE report of the reference plant [2], the initiating events which contribute more than 10% to the total core damage frequency are the

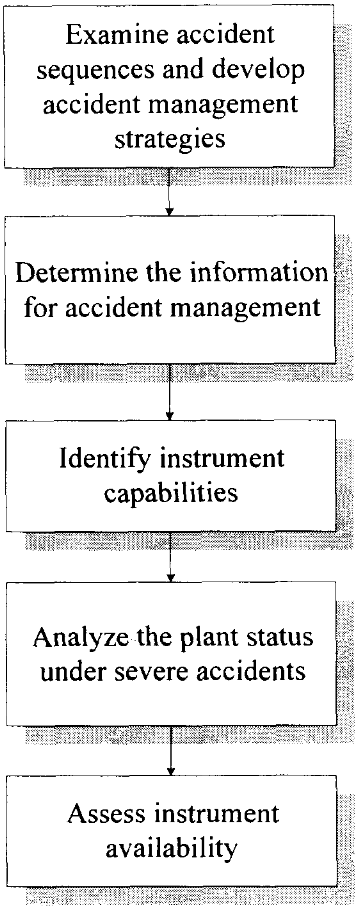

Fig. 1. Task Procedure to Assess Instrument Availability [1]

small-break loss of coolant accident (LOCA) (16.5 %), loss of feedwater (15.1%), station blackout (14.6 %), and loss of a 125V DC bus (10.6%). The initiating events which contribute 5 to 10% to the total core damage frequency are the steam generator tube rupture (SGTR) (8.5%), large-break LOCA (8.4 %), and medium-break LOCA (7.6%). The remaining initiating events contribute about 18.7% to the total core damage frequency.

The containment failure modes used in this IPE report are: containment bypass, early containment failure, late containment failure, and basemat melt-through. Out of these, the most dominant containment failure mode is the containment bypass(10.4%). There are three mechanisms which contribute to the containment bypass: the SGTR sequences, the temperature-induced SGTR, and the interfacing system LOCAs (ISLOCAs). The second important containment failure mode is the late containment failure (6.3%). The containment eventually fails if the containment heat removal is lost and not recovered. The conditional probability of basemat melt-through is determined to be 4.4%. A dry cavity allows the core-concrete interaction and results in the eventual basemat melt-through. Large amounts of debris in the cavity may form non-coolable geometry which allows core concrete interaction even though it is covered with water. Early containment failure is determined to be the least frequent containment failure mode (0.7%). Containment failure before reactor vessel failure was determined to be the most probable cause of early containment failure. The flooded cavity prolonged the time of reactor vessel failure by the external cooling of the reactor vessel. If the containment heat removal is not available for the flooded cavity, the containment may fail prior to the reactor vessel failure. The temperature-induced RCS break makes the alpha mode containment failure apparently important. The higher probability of an alpha mode containment failure is given for low RCS pressure rather than for the high RCS pressure. The containment failure due to the hydrogen burn and/or the direct containment heating (DCH) is determined less important than expected.

In order to prevent such mechanisms as described above and to mitigate the consequence, the candidate AM strategies are suggested as follows: feed and bleed, cavity flooding, spray, recombining, igniting, depressurization of the RCS, isolation, filtering, venting, and so on. These strategies will be matched to mechanisms in the process of constructing the SOTs.

3.2. Information Needs for Accident Management

The SOT is used to determine information needs required for severe accident management. As described in section 2, SOT is constructed as hierarchical structure composed of safety objective, safety functions, challenges, mechanisms, and strategies. Three safety objectives are determined to prevent the reactor vessel failure, to prevent the containment failure, and to mitigate the FP release. The structure and components of each SOT are described below.

(a) *SOT for preventing the reactor vessel failure*

This SOT is constructed for preventing the reactor vessel failure as shown in Figure 2. This tree relates the safety objective to AM strategies for preventing core degradation and confining the melted core within the reactor vessel. These strategies are more effective than those preventing containment failure because much uncertain accident phenomena exist after the melted core has ejected from the reactor vessel out to containment. This safety objective is attained by maintaining two safety functions : one is to maintain the RCS heat removal, and the other is to maintain the reactor vessel boundary.

There are two challenges threatening the safety function for maintaining the RCS heat removal : one is inadequate secondary heat removal, and the other is inadequate primary heat removal. The mechanisms causing the inadequate secondary heat removal may be classified into three categories. The first is inadequate secondary water inventory, which results from the failure of secondary-side feedwater system or inadequate water inventory. AM strategies identified for this mechanism are to develop a secondary feed method and to prepare another secondary feedwater inventory. The second is inadequate secondary pressure control. AM strategy for this mechanism is to bleed the secondaryside by opening the valves to discharge the secondary-side steam. The third is inadequate RCS energy transport which results from the failure of the reactor coolant pump (RCP) or natural circulation failure when high temperature gas is generated from the core. AM strategies for this mechanism are to restart the RCP, or to supply the RCS with water and to ensure water inventory.

The mechanisms causing inadequate primary heat removal may be divided into three groups. The first is inadequate primary water inventory, and AM strategy for this mechanism is to feed the RCS. The second is inadequate primary pressure control, and AM strategy is to bleed the primary side by opening the valves such as PORV (Power Operated Relief Valve). The third is inadequate power control which results from the mechanical failure of the control rod drive system, or from the insertion of inadequately diluted borated coolant, or from the relocation of control rod when core melts down. AM strategies for this mechanism are to develop alternative strategies for control rod insertion and boron insertion, and ensure borated inventory.

After the reactor core and structures have already relocated, the AM objective should be focused on confining relocated materials within the reactor vessel. There are two challenges threatening the safety function for maintaining the reactor vessel boundary : one is the reactor vessel overtemperature, and the other is the reactor vessel overpressure. The mechanisms causing the reactor vessel overtemperature are classified into inadequate core coolability and inadequate reactor vessel water inventory. When the core is relocated in a non-coolable geometry, primary bleed and feed and cavity flooding strategy are recommended. In case water inventory inside the reactor vessel is not sufficient for core cooling when the core is relocated in a coolable geometry, primary feed strategy is recommended.

The mechanism related to the reactor vessel overpressure is due to inadequate reactor vessel pressure control. AM strategy for this mechanism is the primary bleed operation.

(b) SOT for preventing the containment failure

This SOT is constructed for preventing containment failure as shown in Figure 3. In order to achieve the safety objective of preventing the containment failure, three safety functions should be

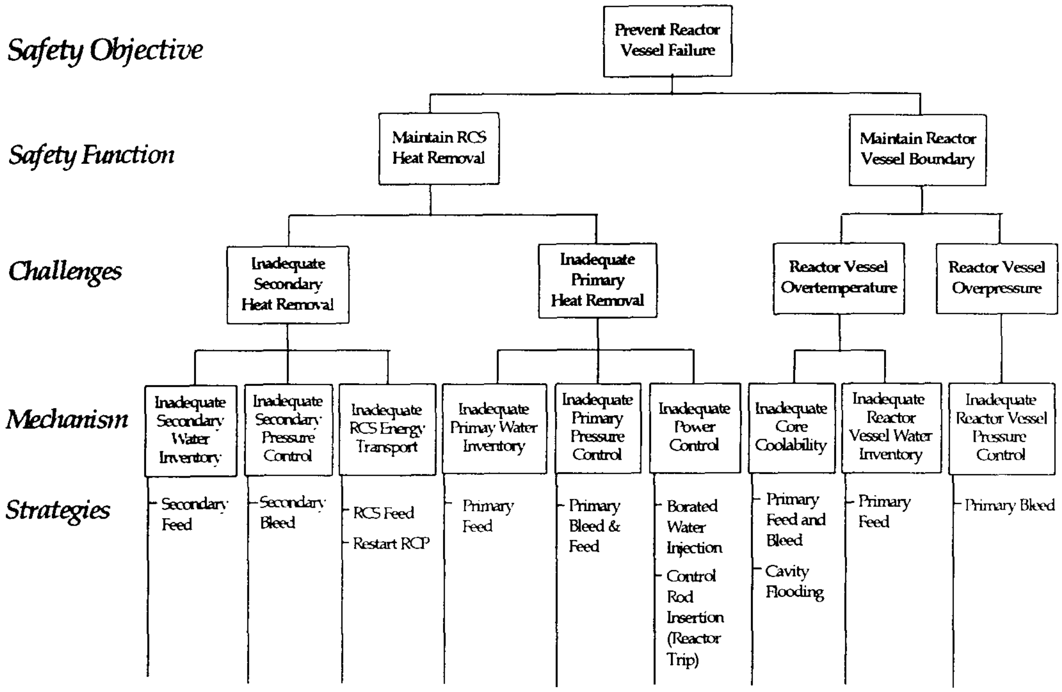

Fig. 2. SOT for Preventing Reactor Vessel Failure

maintained as follows : the first is to maintain pressure control to prevent containment failure from overpressurization, the second is to maintain temperature control to prevent containment failure from overtemperature, and the third is to maintain containment integrity to protect the containment from the internally generated missiles.

There are two challenges threatening the safety function of maintaining the pressure control. One is named as slow pressurization, which increases the containment pressure slowly. The other is quoted as rapid pressurization, which causes rapid pressurization of the containment. The mechanisms causing slow pressurization are due to insufficient heat removal inside the containment and buildup of non-condensable gases. AM strategies for preventing insufficient heat removal are operation of the fan cooler, spray and venting, and those for preventing buildup of non-condensable gas are installation of recombiner and igniter, and venting. The mechanisms causing rapid pressurization are classified into four : DCH, combustible gas detonation, steam explosion, and energy addition at the time of vessel failure. AM strategies for preventing DCH or mitigating the consequence are depressurization of RCS, containment venting, cavity flooding, and adding barriers. For preventing combustible gas detonation, installation of recombiner and igniter, and containment venting strategy are recommended. For preventing steam explosion, it is suggested to eliminate water and to add barriers. For preventing containment heating from the ejected high temperature steam and water, it is recommended to operate the fan cooler system, spray, and venting system.

There are two mechanisms which overtemperature may cause : one is failure of penetration or shell induced from containment overtemperature, and the other is basemat melt-through induced from the interaction between the ejected high temperature core material and cavity concrete. AM strategies for preventing temperature increase due to the penetration or shell failure are to operate the fan cooler

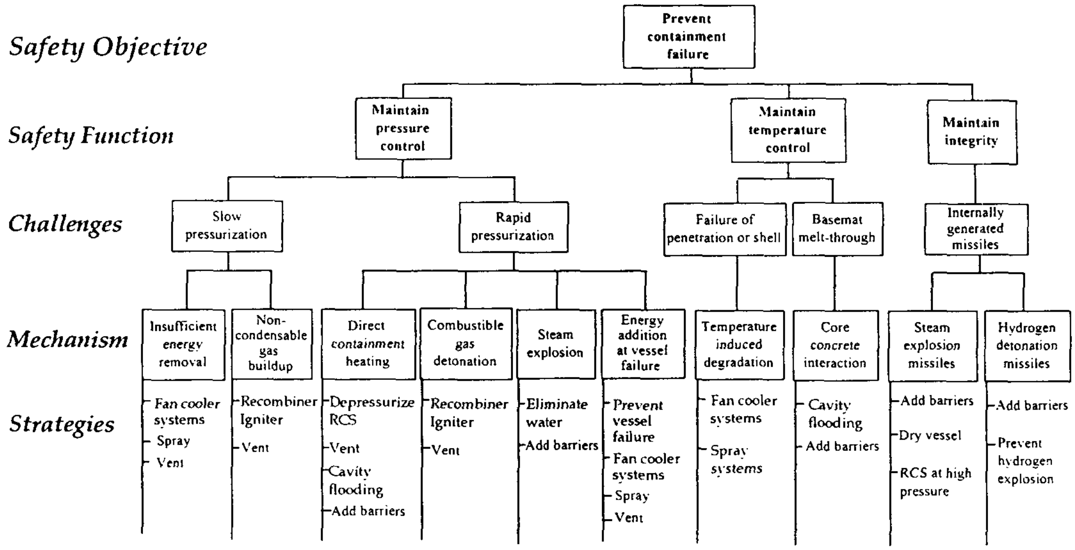

Fig. 3. SOT for Preventing Containment Failure

and spray systems. And for preventing basemat melt-through, cavity flooding or adding barriers are considered as potential strategies.

The third safety function is to protect the containment from the internally generated missiles. The missiles may be generated from the steam explosion or hydrogen detonation missile. The AM strategies for preventing and mitigating the steam explosion missiles are to add barriers and to dry the reactor vessel. Adding barriers, operating the recombiner and igniter, and containment venting are considered to prevent hydrogen detonation.

(c) SOT for mitigating the FP release

The SOT may be built to mitigate the fission product release as shown in Figure 4. This SOT is aimed at relating the safety objective of mitigating the FP release to AM strategies. There are three safety functions to be maintained to achieve the safety objective. The first is to maintain control of FP dispersion to protect FP release out of the containment through the mechanisms of isolation failure, SGTR, and ISLOCA. The second is to maintain control of FP inventory in the atmosphere to prevent the mechanisms of aerosol dispersion and gaseous dispersion. The third is to maintain control of FP release from the containment water to prevent the mechanisms of too low pH of water, hydrolysis, and excessive water temperature.

In order to prevent and mitigate the isolation failure mechanism, reisolation, venting, and operating spray systems are identified as potential strategies. To mitigate the effect of SGTR, ISLOCA mechanisms, repressurization of RCS and flooding the break location are identified. AM strategies for minimizing aerosol FP generation and dispersion are operation of the spray systems and filter system, and chemical reaction. And, for preventing gaseous FP dispersion, chemical reaction and cryogenic system are considered. In order to change acid water into base, adding base water and dilution of acid water, and to prevent hydrolysis, dilution strategy, and to prevent FP release induced from hot water temperature, cooling water system are considered as AM strategies, respectively.

(d) Identification of information needs

After constructing all three SOTs, information needs for AM should be identified. All the information is intended to monitor whether safety

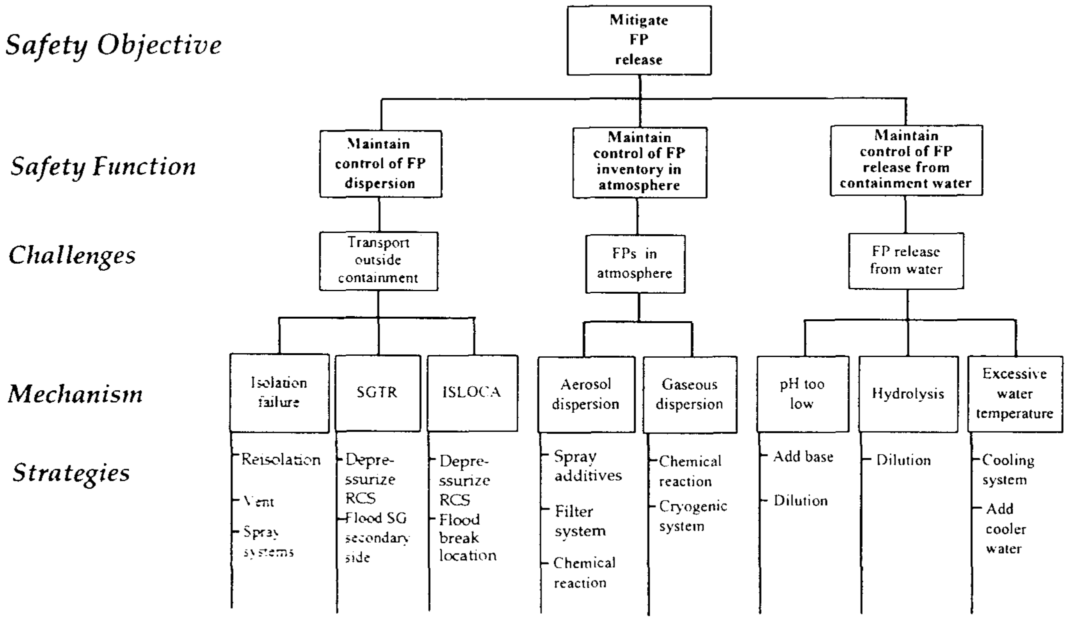

Fig. 4. SOT for Mitigating Fission Product Release

functions are maintained within the safety limit, to detect challenging mechanisms threatening the safety function, and to diagnose the plant and system conditions to initiate the identified AM strategy properly, to observe whether the implemented action is executed and has an effect on the plant as expected. The information needs are examined for the safety objective of preventing the reactor vessel failure among the three safety objectives. Table 1 shows part of the results.

3.3. Instrument Capability

This section presents the capability of instruments installed in the YGN 3&4 nuclear power plants. These include the examination of the provision of instrumentation required by the information needs, and of measurement range and environmental qualification limit of the instruments [3,4]. The measurement ranges of instruments which are considered to be important for AM and installed in YGN 3&4 are shown in Table 2. All the ranges are based on the design requirement of the instrumentation system.

There are information needs which are indispensably required in diagnosis of the plant status and managing the accidents, but are not currently obtainable from the plant. Such types of information needs are listed below. These information needs were identified in the process of assessing the relevant physical phenomena that may threaten the YGN3&4 reactors. These items will provide the operators with a critical piece of information on which they can base their decision-making at the time of crisis. For example, the following information on when the molten core starts to move down to the lower plenum and begins to thermally and chemically attack the reactor vessel will give insight as to the timing and selection of either the in-vessel injection or the cavity flooding or both contingent upon the severity and timing of the situation.

- Information on the core relocation status

If the heat generated in the core is not removed adequately, core temperature increases and fuel, cladding, control rods, and supporting structures may

Table 1. Sample Information Needs Table

	Information Needs	Information Parameters	Existing Instruments
Maintain RCS Heat Removal Safety Function	Heat removal rate	RCS fluid temperature	Hot or cold leg temperature
		RCS pressure	Pressurizer pressure
			RCS pressure
		Steam generator steam flow	Steam flow indicator
		RHR heat removal	RHR flows, temperatures
Inadequate Secondary Water Inventory Mechanism	*Indicator*	Steam generator water level	Steam generator level
	Secondary water inventory		
	Precursor		
	Feedwater flow status	Feedwater flow rate	Main feedwater flow
			Aux. feedwater flow
Secondary Feed Strategy	*Selection Criteria*		
	Inventory availability	Tank inventory	Tank (condenser) level
	Pump capability	Power availability	
	Alignment capability	Steam availability	Valve position indicator
		Valve alignments	
	Strategy Initiation		
	Feedwater flow status	Feedwater flow rate	Main feedwater flow
			Aux. feedwater flow
	Injection water inventory	Tank inventory	Tank (Condenser) level
	Strategy Effectiveness		
	RCS fluid temperature	RCS fluid temperature	Hot or cod leg temperature
	Secondary fluid inventory	Steam generator water level	Steam generator level

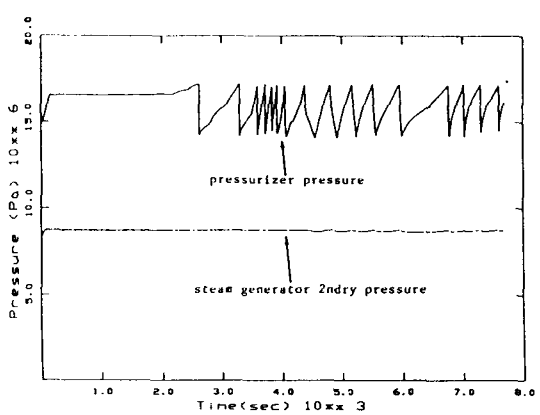

Fig. 5. Pressurizer Pressure and SG Pressure Under SBO Sequence in YGN 3&4 Plants

melt down and relocate to the lower plenum of the reactor vessel. The information on core relocation status would provide the operators with appropriate AM strategy initiation timing and selection criteria whether they initiate the AM strategy for preventing the reactor vessel failure or that for preventing the containment failure. There is no direct instrumentation providing this information. They can only estimate the core relocation status indirectly using the neutron flux detector.

- Information on the lower plenum status

It is possible to predict the reactor vessel failure timing with the help of information on how much core materials have relocated down to the lower plenum. This information is critical to preparing the mitigating strategy to prevent the containment failure. It is suggested that temperature instrument or charge coupled device (CCD) camera should be installed on the exterior wall of the reactor vessel to observe the lower plenum behavior of the reactor vessel.

- Information associated with the cavity flooding

Table 2. The Measurement Ranges of Instruments in YGN 3&4

No.	Available Instruments	Measurement Range
1	Hot leg temperature	0~400℃
2	Cold leg temperature	0~400℃
3	Core exit temperature	Display : 0~1260℃ Recorder : 50~1300℃
4	Pressurizer pressure	0~3,000 psia
5	RCS pressure	0~4,000 psig
6	Pressurizer level	0~100%
7	Reactor coolant level	0~100%
8	Steam generator level	0~100%
9	Steam generator pressure	0~1524 psia
10	S/G total feedwater flow	0~1508 cmH2O
11	S/G downcomer feedwater flow	0~5.5E 5kg/hr
12	Main steam flow	0~3964.8cmH2O
13	Aux. feedwater flow	0~110% design flow
14	Charging flowrate	0~158.5gpm
15	RCP speed sensor	0~1320rpm
16	SIT level	0~100%
17	RWT level	0~100%
18	HPSI flow rate	0~660.5gpm
19	LPSI flow rate	0~6,605gpm
20	Pressurizer safety valve position indicator	Closed/Not closed
21	Boric acid charging flow	0~100% design flow
22	Condensate storage tank level	0~100%
23	Containment spray flow	0~110% design flow
24	Containment sump level	0~100%
25	SDS line temperature	25~175℃
26	SDS line pressure	0~3,000psia
27	SDS Isolation valve	Open/closed
28	ADV	0~100%
29	Charging flow rate	0~158.5gpm
30	Neutron flux power level	0~200% power
31	Containment spray flow	0~20,000l/min

strategy

Cavity flooding is regarded as a useful strategy for maintaining the reactor vessel integrity and for preventing the core-concrete interaction after the reactor vessel failure. Since this strategy has been verified by the experiments and analyses [5,6], supporting instruments for implementing this strategy need to be installed. Necessary instruments are intended to provide information on the cavity flooding system, power availability, water inventory, and cavity water level and temperature to observe the strategy initiation and effectiveness.

On the other hand, all the instruments installed in YGN 3&4 are designed to satisfy the environmental design guide (EDG) which is written based on the analysis result of DBA events for seven local areas (the containment building, auxiliary building, turbine building, control room, main steam isolation valve room, fuel building, and the outside area). Among the seven areas, we list only three areas : the containment building, auxiliary building, and turbine building.

- Containment building
  Maximum temperature : 370°F
  Maximum pressure : 60 psig
  Relative humidity : Saturated/superheated steam & air mixture
  Radiation : 3.3×10^7(40 Yr. TID Rads Plus LOCA)
  Chemical spray : 4,400 ppm Boron as H3BO3
  50 ppm Hydrazine as N2H4
  pH of 7.0~8.5 after 4 hours using Trisodium phosphate
- Auxiliary building
  Maximum temperature : 104°F
  Maximum pressure : Atmosphere
  Relative humidity : 20~90%
  Radiation : 1×10^6 Gamma(accessible areas and all I&C equipment)
  1.25×10^7 Gamma(VCT)
  2.7×10^9 Gamma(Purification Ion Exchanger)
  Chemical spray : Not applicable
- Turbine building
  Maximum temperature : 330°F, 3 minutes
  120°F, 0 to 4 hours
  60°F~104°F, continuous
  Maximum pressure : 3 psig, 3 minutes
  Atmosphere, continuous
  Relative humidity : 95 %
  Radiation : blank

Chemical spray : Not applicable

3.4. Analysis of Severe Accident Conditions

This section shows the results of analysis of the station blackout (SBO) sequences using the SCDAP/RELAP5 code. Three important parameters are examined : pressurizer pressure, steam generator (SG) secondary side pressure, and fuel cladding temperature. These results are compared against the instrument capability identified in section 3.3.

Table 3 shows the event scenario and the occurrence time after the SBO initiates. If the SBO transient starts, all SG feedwater pumps stop, boiling occurs on the SG secondaryside, and the SG water level starts to decrease. Since the turbine stop valve closes and condenser steam dump valve does not function on account of the power loss, the SG pressure begins to increase. When the SG pressure increases over the main steam safety valve (MSSV) setpoint, the SG MSSV opens and SG steam is dumped to the atmosphere, and the pressure is maintained at a nominal value. The SG water level decreases to eventually dryout at 2,835 sec and the SG can no longer function as a heat sink. Since the SG can no longer remove the decay heat generated in the core, the RCS pressure keeps increasing to 17.2 MPa, which is the safety relief valve (SRV) setpoint, at 2,625 sec. As SG loses its function of heat sink completely, the RCS coolant temperature increases steadily to the saturation point of 17.2 MPa at 2,845 sec. The reactor core starts to uncover and fuel cladding temperature increases starting from 5, 340 sec. Afterwards, the fuel cladding temperature rises up to 1,000 K, at which point fuel cladding oxidation begins at 5,880 sec. When the fuel temperature increases, fission products in the fuel elements emerge out to the gap between the fuel and cladding, and the internal pressure of fuel rod increases. Then, as the pressure difference between the inside and outside of the fuel rod increases and ductility increases due to temperature rise, fuel cladding begins to balloon. At 6,364 sec, ballooning arrives at the critical point, and the fuel cladding ruptures. If the fuel cladding temperature rises over 1,700 K, fuel cladding oxidizes rapidly and, due to oxidation heat, fuel cladding temperature increases more quickly. At 6,490 sec, the fuel cladding temperature exceeds 2, 500 K, the oxidized cladding melts, ruptures and relocates down with the melted fuel elements to form cohesive debris bed. At 6,771 sec, the core completely dries out, and at 7,408 sec, the cladding temperature increases to 2,960 K and the oxidized cladding melts down. Upper part of fuel rod materials melts down and relocates to form melting pool and crust on the border of the molten pool. As the temperature rises due to the decay heat generated in the molten pool, crust will breach at 7,607 sec. With crust breaking, materials inside the molten pool slumps down to the lower plenum of the reactor vessel.

Figure 5 shows the variation of the pressurizer and SG secondary side pressures under the SBO transients. At the initial stage, the RCS temperature

Table 3. The Event Scenario and Time During SBO Sequence

Event	Time (sec)
Steady state	0.0
Transient initiated	5.0
Reactor and RCP tripped	5.0
Pressurizer SRV initial opening	2,625.0
Loss of effective heat sink	2,835.0
Hot legs reached saturation temp.	3,845.0
Pressurizer water solid	4,350.0
Natural circulation ended	4,435.0
Core heat up began	5,340.0
Fuel rod clad. oxidation began	5,880.0
Cladding failure by over-strain	6,364.0
ZrO2 rupture and relocation	6,490.0
Core fully dryout	6,771.0
ZrO2 melting	7,407.8
Crust failure in the molten pool	7,607.0
Blockage formation fully in the middle channel of core	7,686.0
Calculation terminated	7,686.0

and pressure are decreasing due to reactor trip and power drop. Since the SG feedwater system stops and steam line to turbine is blocked, the SG pressure rises up to the MSSV setpoint, and the temperature also increases. As the secondary side temperature rises, heat transfer rate from the RCS to the secondary side decreases, and therefore the RCS pressure rises up to the peak point of 16.1 MPa at 250 sec. At about 2,500 sec, as a large portion of the SG U-tubes gets uncovered and heat transfer to the secondary side decreases, the pressurizer pressure rises again. The pressurizer pressure arrives at the SRV setpoint, 17.2 MPa, and the SRV functions to open. Afterwards, the pressurizer SRV continues to repeat opening and closing according to the variation of the pressurizer pressure. The SG pressure also maintains its steady value after once rising up over the MSSV setpoint.

Figure 6 shows the variation of the fuel cladding temperature in the central channel. In this figure, nodes 1, 3, 5, 7, and 9 represent the positions of 0. 32 m, 1.59 m, 2.38 m, 3.02 m, and 3.65 m respectively, from the core bottom. Until the RCS pressure arrives at the SRV setpoint, fuel cladding temperature is maintained at 600 K as the coolant temperature. From 5,340 sec, as core uncovers and upper parts of the core are filled with steam, fuel cladding temperature begins to increase. At about 5,880 sec, fuel cladding temperature arrives at 1,000 K, at which temperature of the fuel cladding starts to oxidize. Because of the heat generation due to oxidization of the fuel cladding, the temperature rises up to over 2,500 K at 6,490 sec. At 6,771 sec, the core becomes completely dried out and the temperature at node 1 escalates quickly. At 7,408 sec, the fuel cladding temperature exceeds 2,960 K, and the oxidized cladding melts down.

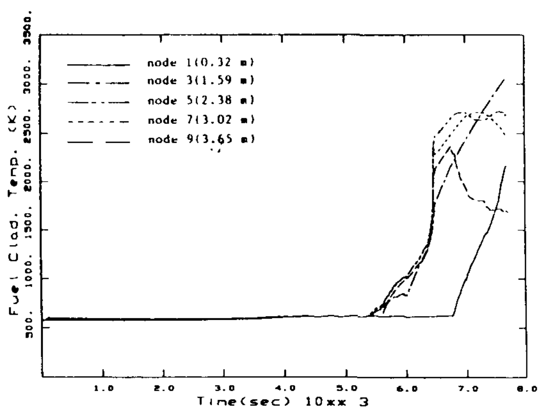

Fig. 6. Fuel Cladding Temperature Under SBO Sequence in YGN 3&4 Plants

3.5. Instrument Availability

This section shows the results of comparison between the instrument capability in section 3.3 and the plant parameters in section 3.4 so as to assess whether existing instruments in YGN 3&4 can provide the information needed for AM. Physical variables analyzed in section 3.4 are the pressurizer pressure, SG secondary-side pressure, and fuel cladding temperature, under the SBO transients. Corresponding instruments of each variable are the pressurizer pressure, SG pressure, and core exit thermocouple. Even though the core exit thermocouple shows somewhat lower value than the fuel cladding temperature, it seems there is not much difference in assessing the instrument availability. The measurement ranges of the above instruments are as follows :

- Pressurizer pressure : 0~3,000 psia (0~20.7 MPa)
- SG pressure : 0~1524 psia(0~10.5 MPa)
- Core exit thermocouple : 0~1573 K

Compared with the variation of the pressurizer pressure and the SG pressure as shown in Figure 5, two corresponding instruments seem to be available under even severe accident conditions, if only the SRV functions properly. On the other hand, in order to use the RCS bleeding strategy, SDS (Safety Depressurization System) valve should be opened. However, there is no dedicated power to open the SDS valve in the SBO event. And, it is required to prepare a dedicated power supply system or DC valve to control the SDS valve in the SBO event.

For fuel cladding temperature, it goes over 1,600 K after 6,300 sec, and this temperature exceeds the measurement range of the core exit thermocouples. Therefore, the core exit thermocouples seem to be available only before the fuel elements begin to melt down.

4. Conclusions

According to the task procedure for the assessing the instrument availability, the SOTs have been constructed for preventing the reactor vessel failure, preventing the containment failure, and mitigating the FP release. We determined information needs and listed existing instruments that the operators will need to prevent the reactor vessel failure in the YGN 3&4 plants. We also examined existing instrument capability, analyzed the plant status during the SBO sequences to assess the instrument availability under severe accident conditions. We obtained the following results from this study:

1. Additional information is required to diagnose the core relocation and reactor vessel lower plenum status after core has melted. These instruments are indispensable for the operators to make a decision which AM strategies be adopted.
2. In association with the cavity flooding strategy, information on the cavity flooding system, cavity water level and cavity temperature should be provided to initiate cavity flooding, and to observe the strategy execution and effectiveness.
3. The temperature measuring instruments become no longer available after the core relocation. The pressure instruments are to some extent available during all the SBO sequences. Therefore, it is necessary to install new temperature instruments providing a maximum temperature value over all of the sequences. As an alternative method, the AM plan for terminating the accident progression before core relocation should be strengthened.
4. In order to effectuate the RCS bleeding strategy during a SBO event, it is required that SDS valve be opened by connecting dedicated AC or DC power.

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