RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

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Question No. 07.05-6

Clarify why the APR1400 has no Type A variables when there are manual actions described in FSAR Tier 2, Chapter 15, Section 15.0.0.6.

10 CFR Part 50, Appendix A, General Design Criteria 13, "Instrumentation and Controls," requires, in part, instrumentation to be provided to monitor variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and accident conditions. RG 1.97, Rev. 4 endorses IEEE Std 497-2002. IEEE Std 497-2002, Section 4.1, states "Type A variables are those variables that provide the primary information required to permit the control room operating staff to:

> a) Take specific planned manually-controlled actions for which no automatic control is provided and that are required for safety systems to perform their safety-related functions as assumed in the plant Accident Analysis Licensing Basis
>
> b) Take specific planned manually-controlled actions for which no automatic control is provided and that are required to mitigate the consequences of an AOO.
>
> Type A variables provide information essential for the direct accomplishment of specific safety- related functions that require manual action. These variables are a subset of those necessary to implement the plant specific emergency procedure guidelines (EPGs) or the plant specific emergency operating procedures (EOPs) or the plant abnormal operating procedures (AOPs)."

In RAI 38-7878, Question 07.05-2, the applicant indicated there were no Type A variables. However, in Chapter 15 of the APR1400 FSAR Tier 2, the staff finds manual actions are credited for certain safety functions, such as those to address a steam generator tube rupture. Using the methodology above as stated in IEEE Std 497-202, provide a table of all operator actions required by Chapter 15; listing all indications that the operator would rely on, the location and qualification for each indication that is needed by the operator. Provide the basis

for why there are no Type A variables when it appears in FSAR Tier 2, Chapter 15, that manual operator actions are credited to address design basis events.

Response

DCD Tier 2, Chapter 15, "Transient and Accident Analyses" was reviewed and manual actions associated with event mitigation have been tabulated, as shown in Table 1.

Table 1. Manual Actions for Accident Analyses

	Title of Event	Frequency	Manual Actions
15.1	Increase in Heat Removal by the Secondary System		
15.1.1	Decrease in Feedwater Temperature	AOO	None
15.1.2	Increase in Feedwater Flow	AOO	None
15.1.3	Increase in Steam Flow	AOO	None
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	AOO	Manual reactor trip Isolation of inadvertent open valve Plant cooldown
15.1.5	Steam System Piping Failure Inside and Outside the Containment (MSLB)	PA	Plant cooldown Termination of AFW (EOG)
15.2	Decrease in Heat Removal by the Secondary System		
15.2.1	Loss of External Load	AOO	Plant cooldown
15.2.2	Turbine Trip	AOO	Plant cooldown
15.2.3	Loss of Condenser Vacuum	AOO	Plant cooldown
15.2.4	Closure of the Main Steam Isolation Valve	AOO	Plant cooldown
15.2.5	Steam Pressure Regulator Failure	N/A	N/A
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	AOO	Plant cooldown
15.2.7	Loss of Normal Feedwater Flow	AOO	
15.2.8	Feedwater System Pipe Break Inside and Outside the Containment	PA	Plant cooldown
15.3	Decrease in Reactor Coolant System Flow Rate		
15.3.1	Loss of Forced Reactor Coolant Flow	AOO	Plant cooldown
15.3.2	Flow Controller Malfunctions	NA	N/A
15.3.3	Reactor Coolant Pump Rotor Seizure	PA	Plant cooldown
15.3.4	Reactor Coolant Pump Shaft Break	PA	Plant cooldown
15.4	Reactivity and Power Distribution Anomalies		
15.4.1	Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low-Power Startup Condition	AOO	Plant cooldown
15.4.2	Uncontrolled Control Element Assembly Withdrawal at Power	AOO	None
15.4.3	Control Element Assembly Misoperation	AOO	None
15.4.4	Startup of an Inactive Reactor Coolant Pump	AOO	None
15.4.5	Flow Controller Malfunction	N/A	N/A
15.4.6	Inadvertent Decrease in Boron Concentration in the Reactor Coolant	AOO	Turn off charging pump Increase RCS boron

	System		concentration
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	AOO	None
15.4.8	Spectrum of CEA Ejection Accidents	PA	Plant cooldown
15.5	Increase in Reactor Coolant Inventory		
15.5.1	Inadvertent operation of emergency core cooling system (ECCS) that increases reactor coolant inventory	AOO	None
15.5.2	Chemical and volume control system malfunction that increases reactor coolant inventory	AOO	Plant cooldown
15.6	Decrease in Reactor Coolant Inventory		
15.6.1	Inadvertent opening of a pressurizer pressure relief valve		See 15.6.5
15.6.2	Failure of small lines carrying primary coolant outside the containment	AOO	Manual reactor trip Isolate broken line
15.6.3	Steam generator tube failure (SGTR)	PA	SG isolation Plant cooldown using unaffected SG Termination of SI (EOG)
15.6.4	BWR event	NA	N/A
15.6.5	Loss-of-coolant accidents (LOCAs) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (RCPB)	PA	LBLOCA : simultaneous hot leg and cold leg injection to flush boric acid from the vessel. SBLOCA : plant cooldown
15.7	Radioactive Material Release from a Subsystem or Component		
15.7.1	Radioactive gas waste system leak or failure	PA	None
15.7.2	Radioactive liquid waste system leak or failure	NA	None
15.7.3	Postulated radioactive releases due to liquid-containing tank failures	PA	None
15.7.4	Fuel handling accident	PA	None
15.7.5	Spent fuel cask drop accident	PA	None
15.8	Anticipated Transient without Scram	Beyond DBE	

Indications to take the manual actions, indication location, indication qualifications, and the basis for AMI selection for each manual action are discussed in Table 2 and the notes thereto.

Table 2. Basis for AMI Type A selection

Title of Event	Manual Actions	Indication to Take Manual Action	Location/ Qualification	Basis for AMI Selection
15.1.4 IOSGADV	Manual Rx trip	Logarithmic Reactor Power SG pressure	MCR/1E	Note 1
	Isolate IO valve	Valve position indications	MCR/1E/non-1E	Note 2
	Plant Cooldown	-	-	Note 3
15.1.5 MSLB	Termination of AFW (EOG)	SG pressure and water level	MCR/1E	Note 4

15.4.6 Boron Dilution	Turn off charging pump Increase RCS boron concentration	Logarithmic reactor power Reactor makeup flow Boric acid flow	MCR/1E MCR/non-1E MCR/non-1E	Note 5
15.4.8 CEA Ejection	Plant cooldown	-	-	Note 3
15.5.2 CVCS Malfunction	Plant cooldown	-	-	Note 3
15.6.2 LDLB	Manual reactor trip Isolate broken line	Letdown line pres. PZR level	MCR/non-1E MCR/1E	Note 6
15.6.3 SGTR	SG isolation	NA		Note 7
	Plant cooldown using unaffected SG	Radioactivity detector SG water levels	MCR/non-1E MCR/1E	Note 7
	Termination of SIS (EOG)	SIS termination criteria	MCR/1E	Note 8
15.6.5 LOCA	Hot leg injection	Time	NA	

Note 1. During an inadvertent opening of a secondary system valve, reactor power stabilizes at 113% due to increased steam flow.

A manual reactor trip is required to terminate the transient. However, a reactor trip is not necessary to meet the safety analysis acceptance criteria. The reactor trip is assumed to check if fuel failure occurs at the time of the reactor trip.

Also, as described in Tier 2 Chapter 15, Table 15.0-7, there are available reactor trip functions for AOO. These reactor trip functions can be considered as automatic functions provided.

Therefore, the manual reactor trip is not selected as a Type A variable for operator action.

Note 2. **Restoring Event Initiating Cause**: Position indications on the secondary valves are necessary to identify IO valve position. This is applicable for all SGADVs, MSSVs, and turbine bypass valves. KHNP does not believe the intent of the AMI requirements is to make all the valve positions 1E indications and Type A variables.

Plant parameters for event diagnosis are considered different from the accident monitoring variables for operator actions. Event diagnosis should be performed with all available plant indications. Type A plant parameters for operator actions are primary variables for an operator to perform specific actuations on safety systems by monitoring if the variables will reach specified setpoints after the cause of the initiating event is diagnosed. “Planed action” means “planed action for a certain diagnosed accident.” Therefore, Type A variables are not necessary for restoring the cause of an initiating event.

For a secondary side steam release event, the radiation release limit is not challenged.

Note 3. **Initiating Plant Cooldown**: The safety analysis in Tier 2 Chapter 15 concludes that when an operator initiates plant cooldown, the plant can be brought to a safe condition. In most of the events, plant parameters are in a controlled state (not changing in an adverse direction) before the operator initiates plant cooldown. In a few events, such as CEA ejection (15.4.8), CVCS malfunction (15.5.2) and SGTR (15.6.3), MSSV cycling operation continues until the operator initiates plant cooldown using SGADV(s).

The initiation of plant cooldown is to ensure the heat removal function. An operator is required to take action to initiate cooldown according to the APR1400 Emergency Operation Guide (EOG) even without diagnosis of the accident. Therefore, there is no need to specify Type A variables for initiating plant cooldown.

Note 4. **Termination of Auxiliary Feedwater** (AFW) is required as an operator action to mitigate the Main Steam Line Break (MSLB) accident in the APR1400 EOG. For the APR1400 MSLB analysis, auxiliary feedwater is assumed to be provided to both steam generators. The peak reactivity occurs at about 400 seconds and then it gradually decreases. Therefore, the isolation of auxiliary feedwater is not a required operator action to meet the Tier 2, Chapter 15 safety analysis acceptance criteria.

Also, each AFW storage tank has 100% capacity for DBEs, and there is no need to control the inventory of the tank. Termination of AFW to a faulted SG is not required for any of the analyzed events:

- A steam generator tube rupture,
- An unisolatable main feedwater line break,
- An unisolatable steam generator blowdown line break,
- An unisolatable main steam line break, and
- A stuck open main steam safety valve.

Also, as discussed in Note 8, the “termination of safety system” is not considered to conform to the Type A definition of a variable which enables “safety systems to perform their safety-related functions.” The safety-related function of the AFWS is to provide SG water inventory for safe shutdown.

However, since termination of AFW is important to prevent core overcooling, Type A variables are selected for the termination of AFWS. Steam generator pressure and water level are primary variables to diagnose MSLB and Main Feedwater Line Break.

Note 5. The corrective action for the boron dilution event is to restore the event initiating cause. Refer to Note 2. Boron dilution can be caused by failure of the makeup control system or charging system misalignment. Any specific operator action cannot be pre-planned.

Operator action to terminate boron dilution event is necessary to protect reactor fuel. Therefore, logarithmic reactor power is selected as a Type A variable since this variable is an indication of a boron dilution event.

Note 6. Letdown line break (LDLB) is an AOO event. Safety analysis confirms that the

radiation release meets the acceptance criteria when the operator isolates the break flow at 30 minutes after the initiation of the event. As discussed in Note 2, manual reactor trip is not required to satisfy safety analysis acceptance criteria.

Also, as described in Tier 2 Chapter 15, Table 15.0-7, there are available reactor trip functions for LDLB event. These reactor trip functions can be considered as automatic functions provided.

Operator action to isolate the break flow is necessary to limit the radiation release and restore the plant. A pipe break, such as a letdown line break, charging line break, sampling line break, or instrument tubing line break, will have a similar transient. The operator action for this event is also considered as restoring initiating event causes, as discussed in Note 2.

However, since reactor coolant is directly released to outside of the containment and the radiation release is utmost safety concern, KHNP considers that some variables are to be assigned as Type A variables to facilitate event diagnosis and early initiation of operator action. The selected variables are as follows:

Pressurizer pressure (wide range), and
Pressurizer water level

Pressurizer pressure and water level are primary indication of decrease in RCS inventory.

Note 7. **Steam Generator Tube Rupture (SGTR) :**
DCD Tier 2 describes the SGTR diagnosis and operator actions as follows :

After a reactor trip, the operator begins to cool down the hot leg temperature using the turbine bypass valves to the saturation temperature corresponding to the main steam safety valve (MSSV) opening setpoint. The operator then cools the nuclear steam supply system (NSSS) to shutdown cooling entry conditions using the unaffected SG after isolating the affected SG or verifying that it is isolated.

Diagnosis of the SGTR accident is facilitated by radiation monitors that initiate alarms and inform the operator of abnormal activity levels and that corrective operator action is required. The detectors are installed in the condenser air ejector exhaust, SG blowdown lines, and main steam line. Additional diagnostic information is provided by RCS pressure and pressurizer level responses indicating a leak and by a level response in the affected SG.

The operator can identify the affected SG by radioactivity detector or water level variation after the reactor trip. After identifying the affected SG, the operator isolates the affected SG or confirms that it is isolated. Using the plant emergency procedure, the operator continues to cool the NSSS manually using the operation of the auxiliary feedwater system and the ADVs of the unaffected SG. The analysis presented here conservatively assumes that operator action is delayed until 30 minutes after the first indication of the event.

Since the SGs are automatically isolated by the main steam isolation signal (MSIS),

operator action for isolating SGs is not an operator action requiring Type A variable.

After reactor trip, the operator initiates plant cooldown, as described in Note 3. The APR1400 emergency operating guide (EOG) standard post trip actions (SPTA) require verification of heat removal and operation of the ADV if necessary. Also, the EOG provides the functional recovery guideline (FRG) for core and RCS heat removal. The FRG is a set of immediate operator actions to be taken, even before an event is classified or diagnosed. Therefore, initiating cooldown by opening the ADV may not be considered as an operator action requiring Type A variable.

The safety analysis for the SGTR requires plant cooldown using the unaffected steam generator. If cooldown is performed with the affected SG, radiation release would increase more than assumed in the safety analysis. Therefore, plant parameters for identifying the affected SG may be considered necessary.

APR1400 EOG provides the diagnostic actions (DA) guide. The operator diagnoses the affected SG by following the DA guide. Event diagnosis should be performed with all available plant variables.

“Planned manual action” implies that the action is planned for a certain accident. If an operator diagnosed an SGTR event with all available plant parameters, he already knows which SG is impacted and does not need any indications to open the ADV of the intact SG.

However, since reactor coolant is directly released to the outside of containment and a radiation release is the utmost safety concern, KHNP considers that some variables should be assigned as Type A variables to facilitate event diagnosis including identification of the affected Steam Generator and early initiation of operator action. The selected variables are as follows:

Pressurizer pressure (wide range),
Pressurizer water level,
Steam Generator pressures, and
Steam Generator water level (wide range).

Pressurizer pressure and water level are primary indications of a decrease in RCS inventory. Steam Generator pressure and water level are used for diagnosis of SGTR and identification of an affected Steam Generator.

Note 8. **Termination of Safety Injection (SI)** is required by the SGTR emergency operating guide. If SI is not terminated, RCS pressure maintains high, break flow continues, and the main steam safety valves are opened and release radiation. If SI is not terminated, more radiation release would occur than assumed in the safety analysis.

However, termination of SI can be considered to be included in the EOG plant cooldown process. Variables for SI termination are classified as Type B variables. Furthermore, IEEE Std 497-2002, Section 4.1 states that:

a) *Take specific planned manually-controlled actions for which no automatic control is provided and that are required <u>for safety systems to perform their safety-related</u>*

<u>functions</u> as assumed in the plant Accident Analysis Licensing Basis

The safety-related function of the safety injection system is to make up the core coolant. Therefore, termination of SI does not conform to the IEEE Std 497-2002 Type A definition.

However, since SI termination is critical operator action to terminate break flow, the variables for verifying SI termination criteria are assigned as Type A variables. SI termination criteria are as follows-:

(1) RCS subcooling is greater than or equal to [minimum RCS subcooling],
(2) Pressurizer level is greater than [lower end of post trip pressurizer level target band] and not lowering,
(3) At least one SG is available for RCS heat removal with level being maintained or restored to [post-trip SG level band],
(4) The RVUH level is greater than [top of the hot leg nozzle].

Thus, additional Type A variables are:

RCS hot leg temperature (wide range),
RCS cold leg temperature (wide range),
RCS subcooling margin, and
CET subcooling margin.

RCS or CET subcooling may be evaluated using Pressurizer pressure and RCS or CET temperature and can be excluded. However, during accident conditions, it would be better for an operator to have final calculated values. Pressurizer level and RCS or CET subcooling can be considered as primary variables for RVUH level.

In summary, KHNP has selected the Type A accident monitoring variables in the APR1400 as follows:

Pressurizer pressure (wide range),
Pressurizer water level,
Steam Generator pressures,
Steam Generator water level (wide range),
RCS hot leg temperature (wide range),
RCS cold leg temperature (wide range),
RCS subcooling margin,
CET subcooling margin, and
Logarithmic reactor power

Type A variables have been selected for the APR1400. Because the selected Type A variables are also Type B or C variables and the requirements for Type A instrumentation are the same as those of Types B and C instrumentation, the selection of Type A variables does not impact the current AMI design.

Impact on DCD

DCD Tier 1, Table 2.5.3-2 and DCD Tier 2, Chapter 7 will be revised as indicated in the attachment of the revised response to RAI 38-7878, Question 07.05-1.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

Table 3.3.11 and B 3.3.11 will be revised as indicated in the attachment of the revised response to RAI 38-7878, Question 07.05-1.

Impact on Technical/Topical/Environmental Reports

The Safety I&C System Technical Report, APR1400-Z-J-NR-14001-P and Software Program Manual TeR, APR1400-Z-J-NR-14003-P will be revised as indicated in the attachment of the revised response to RAI 38-7878, Question 07.05-1.