MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU
TOKYO, JAPAN

August 28, 2008

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco,

Docket No. 52-021
MHI Ref: UAP-HF-08155

Subject: MHI's Responses to US-APWR DCD RAI No.40

References: 1) "Request for Additional Information No. 40 Revision 0, SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: PRA," dated July 29, 2008.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Responses to Request for Additional Information No.40 Revision 0".

Enclosed are the responses to the RAIs that are contained within Reference 1.

As indicated in the enclosed materials, this document contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes responses to the RAIs (Enclosure 2) and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all information in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is below.

Sincerely,

Y. Ogata

Yoshiki Ogata,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

D081
NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Responses to Request for Additional Information No.40 Revision 0 (proprietary)

3. Responses to Request for Additional Information No.40 Revision 0 (non-proprietary)

CC: J. A. Ciocco
C. K. Paulson

Contact Information

C. Keith Paulson, Senior Technical Manager
Mitsubishi Nuclear Energy Systems, Inc.
300 Oxford Drive, Suite 301
Monroeville, PA 15146
E-mail: ck_paulson@mnes-us.com
Telephone: (412) 373-6466

ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-08155

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Responses to Request for Additional Information No.38 Revision 0" dated August 2008, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information identified as proprietary in the enclosed document has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the unique design and methodology developed by MHI for performing the design of the US-APWR reactor.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the information contained in the referenced document would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

   A. Loss of competitive advantage due to the costs associated with development of methodology related to the analysis.

B. Loss of competitive advantage of the US-APWR created by benefits of modeling information.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 28th day of August 2008.

Y. Ogata

Yoshiki Ogata,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosure 3

UAP-HF-08155
Docket Number 52-021

Responses to Request for Additional Information No.40 Revision 0

August, 2008
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.40-610 REVISION 0

SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

APPLICATION SECTION: PRA

DATE OF RAI ISSUE: 7/29/2008

QUESTION NO. : 19-86

It is stated (DCD Section 19.1.3 "Special Design/Operational Features") that the residual heat removal system (RHRS) piping is designed to withstand a higher pressure than operating plants. This design feature was used in modeling interfacing systems LOCA in the PRA. It was assumed that following a break of the reactor coolant system (RCS) boundary at the RHRS suction or injection lines, the reactor coolant will flow to the refueling water storage pit (located inside the containment) unless a break occurs at the RHRS piping outside the containment. The probability that a break will occur at the RHRS piping outside containment, given RHRS over-pressurization, was based on a piping rupture rate of 1.5E-10/hr-ft. Even though the RHRS piping is designed to withstand higher pressures than operating plants, this rupture rate is not applicable to piping that may be pressurized above its design capability. Please provide information that justifies the assumed rupture rate for over-pressurized RHRS piping. Also, please summarize the interfacing systems LOCA risk evaluation in Chapter 19 of the DCD by including (1) important results, (2) risk insights regarding the design and operational features which contribute to the low risk associated with interfacing systems LOCA, and (3) key assumptions made in the analysis.

ANSWER:

US-APWR is designed so that the residual heat removal system (RHRS) pressure does not exceed its critical pressure and RHRS break due to overpressure does not occur in case a break of the reactor coolant system (RCS) boundary at the RHRS suction or injection lines happens. It is also designed that pressure rise due to leakage from RCS boundary isolation valve can be mitigated by RHRS relief valves. It is therefore considered appropriate to apply a generic piping rupture rate to RHRS break. Even if the pipe rupture rate is assumed to be 100 times larger considering the severe condition, the frequency of an interfacing system LOCA (ISLOCA) is estimated to be 3E-10/yr. This simple evaluation can conclude that the sensitivity of the RHRS piping rupture rate is insignificant. Overall, the treatment of an ISLOCA for the US-APWR PRA can be summarized as following.

(1) The occurrence frequency of an ISLOCA is evaluated as negligible.

(2) A design Feature that RHRS piping withstands higher pressure than operating plants and that the reactor coolant following a break of the RCS boundary at the RHRS suction or injection lines flows into the in-containment refueling water storage pit (RWSP) can significantly reduce the risk due to an ISLOCA.

(3) One of the key assumptions is that the occurrence frequency of the RCS boundary break at the RHRS suction or injection lines is limited to the same level with operating plants.

Above discussions will be incorporated into the revised technical report [1)] and summarized in the next revision of the DCD.

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

Impact on DCD

DCD will be revised to address the information discussed for this RAI.

Impact on COLA

This RAI and its response will impact the COLA, which refers the DCD.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.40-610 REVISION 0

SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

APPLICATION SECTION: PRA

DATE OF RAI ISSUE: 7/29/2008

QUESTION NO. : 19-87

The initiating event categories, with their respective frequencies, are listed in Table 19.1-2 of the DCD. The frequency of each category is listed without any information about its uncertainty (e.g., error factor). Error factors associated with these frequencies, with the exception of the total loss of component cooling water/emergency service water (CCW/ESW) initiating event frequency, are reported in the references (NUREG/CR-6928 and NUREG/CR-5750) provided in Table 19.1-2 of the DCD. Regarding the total loss of CCW/ESW initiating event frequency, it is not clear whether its error factor was assessed. The error factors reported in NUREG/CR-6928 and NUREG/CR-5750 indicate that the uncertainty in some initiating events may be major contributors to the uncertainty of the estimated risks (e.g., CDF). Please explain how these uncertainties are addressed in the PRA.

ANSWER:

The error factor (EF) of frequency for each initiating event (IE) is summarized in the following table. This table also describes the basis for determining the EF. Large EF (=10) is assumed when there are no past records in NUREG/CR-6928. Otherwise small EF (=3) is assumed with a few exceptions. We confirmed that there are no significant sensitivities on the results of uncertainty analysis even if the EFs used here are different from those reported in NUREG documents.

This information will be included in the DCD next time.

Table Initiating Events for the US-APWR

	IE	Event Description	Frequency	EF	Basis for EF	Reference
1	LLOCA	Large Pipe Break LOCA	1.2E-06	10	There are no past records in NUREG/CR-6928.	NUREG/CR-6928 (Reference19.1-16)
2	MLOCA	Medium Pipe Break LOCA	5.0E-04	10	There are no past records in NUREG/CR-6928.	NUREG/CR-6928
3	SLOCA	Small Pipe Break LOCA	3.6E-03	10	There are no past records in NUREG/CR-6928.	NUREG/CR-6928
4	VSLOCA	Very Small Pipe Break LOCA	1.5E-03	10	Although there are past records in NUREG/CR-6928, large EF is assumed from engineering judgment.	NUREG/CR-6928
5	SGTR	Steam Generator Tube Rupture	4.0E-03	3	There are past records in NUREG/CR-6928.	NUREG/CR-6928
6	RVR	Reactor Vessel Rupture	1.0E-07	-	Because the frequency is low enough, distribution for the frequency is not considered.	WASH-1400 (Reference19.1-22)
7	SLBO	Steam Line Break/Leak (Downstream MSIV : Turbine side)	1.0E-02	10	Because NUREG/CR-6928 contains no information on EF, large EF is assumed from engineering judgment.	NUREG/CR-5750 (Reference19.1-45)
8	SLBI	Steam Line Break/Leak (Upstream MSIV : CV side)	1.0E-03	10	Because NUREG/CR-6928 contains no information on EF, large EF is assumed from engineering judgment.	NUREG/CR-5750

	IE	Event Description	Frequency	EF	Basis for EF	Reference
9	FWLB	Feed-water Line Break	3.4E-03	10	Because NUREG/CR-6928 contains no information on EF, large EF is assumed from engineering judgment.	NUREG/CR-5750
10	TRANS	General Transient	0.8	3	There are past records in NUREG/CR-6928.	NUREG/CR-6928
11	LOFF	Loss of Feed-water Flow	1.9E-01	3	There are past records in NUREG/CR-6928.	NUREG/CR-6928
12	LOCCW	Loss of Component Cooling Water	2.3E-5	10	There are no past records in NUREG/CR-6928.	NUREG/CR-6928
13	PLOCW	Partial Loss of Component Cooling Water	1.2E-3	10	Although there are past records in NUREG/CR-6928, large EF is assumed from engineering judgment.	NUREG/CR-6928
14	LOOP	Loss of Offsite Power	4.0E-2	3	There are past records in NUREG/CR-6928.	NUREG/CR-6928
15	LOAC	Loss of Vital ac Bus	9.0E-3	3	There are past records in NUREG/CR-6928.	NUREG/CR-6928
16	LODC	Loss of Vital DC Bus	1.2E-3	3	There are past records in NUREG/CR-6928.	NUREG/CR-6928

Impact on DCD

DCD will be revised to address the information discussed for this RAI.

Impact on COLA

This RAI and its response will impact the COLA, which refers the DCD.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.40-610 REVISION 0

SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

APPLICATION SECTION: PRA

DATE OF RAI ISSUE: 7/29/2008

QUESTION NO. : 19-88

The total loss of component cooling water/emergency service water (CCW/ESW) initiating event frequency was estimated, by using fault tree analysis (FTA), to be 2.3E-5/yr (Table 19.1-2 of the DCD). This frequency estimate is significantly lower than what is historically used in PRAs for similar operating reactors (e.g., NUREG/CR-6928 recommends a frequency of 8E-4 for the total loss of CCW/ESW initiating event). The staff notes that the US-APWR design has a four CCW/ESW train configuration completely separated into two independent subsystems. However, only two pumps are normally running and one of the two standby trains has no TS outage requirements. The information provided in Section 6A.16 of the PRA (MUAP-07030, Rev 0) is not adequate for the staff to understand how the frequency of total CCW/ESW loss was estimated. Please explain the methodology that was used and state the assumptions that were made in the FTA. Also, please clarify the following: (1) how the mission time of one year (item e on page 6A.16-1) was used in the FTA; (2) the meaning of the 24-hour mission time shown in the list of basic events (Table 6A.16-1); (3) the meaning of CCF events reported in Table 6A.16-2 and their assumed probabilities; and (4) how the basic events reported in Tables 6A.16-1 and 6A.16-2 are related to the basic event identifiers used in fault tree IE-CCW-SWS and in the minimal cut sets.

ANSWER:

According to NUREG/CR-6928, there has been no experience of total loss of CCW/ESW initiating event in the US commercial nuclear industry. The US-APWR is designed to provide two independent subsystems for CCW/ESW as pointed by the staff, and this design feature is considered as one of the advantages of US-APWR against existing plants. The initiating event frequency of this initiating event is therefore expected to be lower than the value recommended in NUREG/CR-6928.

FTA is also recommended as a method to evaluate initiating frequencies as well as statistical process. In the US-APWR PRA, FTA is applied in order to appropriately address the specific design features in the evaluation. In the fault tree, a one year mission time to evaluate the annual frequency of the event is considered for the failures that start the sequence of events that cause the initiating event. 24 hour mission time is considered for secondary failures in the sequence of events that cause the initiating event. The typical 24 hour mission time was judged to be applicable considering time to repair and restore the first failure, and time to achieve stable plant conditions that can be maintained regardless of the availability of CCW.

Answer to the question (1) and (2):

1) It is assumed that 2 trains (A and C) out of 4 CCW/ESW trains are normally operating and trains B and D are in standby.

2) Assuming condition described in 1), a one year (8760 hours) mission time is considered to evaluate the annual frequency of the first failure in trains A and C that start the sequence of events that cause the total loss of CCW/ESW. 8760 hour mission time described in Attachment 6A page 16-1 item "e" and shown in Table 6A.16-1 is this one year mission time.

3) In the fault tree, failures of all four trains of the CCW system and ESW system is modeled as secondary failures in the sequence of events that may cause the initiating event. The mission time of 24 hours described in Table 6A.16-1 represents time to repair and restore the first failure, and time to achieve stable plant conditions that can be maintained regardless of the availability of CCW.

Answer to the questions (3):

Common cause failure (CCF) for normally operating components that start the sequence of events that cause the initiating event is considered as well as CCF of standby components that occur during the 24 hour mission time, and are modeled in the fault tree.

Table 6A.16-2 lists common cause group identifiers and basic event identifiers assigned to each of the common cause groups. For each basic event identifiers, CCF basic events that represent CCF between the basic events assigned in the same basic event group are considered. This is performed by the "CCF-Group" function of the PRA code "Riskspectrum". This function is described in the response to question 19-40 of RAI#25.

Table 19-88-1 provided here shows detail of the CCF events considered in the fault tree. For each CCF group, the table provides information on components considered, description of the CCF, basic event that are assigned to the CCF group, CCF group size, basic event identifiers that represent the CCF event, and the mean value of the CCF probability.

CCF events are quantified using the MGL method described in Chapter 8 of the PRA technical report. Exceptions are CCF events of component of asymmetric configuration, which is CCF between standby components and running components. For such CCF events, the “CCF-Group” function is not applied but a basic event with a identifier shown in the fourth column of Table 6A.16-2 is modeled assigned with a probability of the CCF event. The method applied to quantify CCF of components in asymmetric configuration is described in section 8.7, Chapter 8 of the PRA technical report.

Answer to the questions (4):

CCF events that are quantified using the “CCF-Group” function of the PRA code “Riskspectrum” are appears in the MCS as basic events starting from common cause group identifier shown in Table 6A.16-2 and followed by characters such as “-ALL”. These basic events identifiers appear in the MCS. Detail of the naming rule of CCF basic events is described in the response to question 19-40 of RAI#25. These basic events do not appear in the fault tree, but are logically located in the same place with the basic events assigned to CCF group. This is also described in response to question 19-40.

For CCF of asymmetric configuration, the basic event identifiers shown in the fourth column of Table 6A.16-2 are model in the FT as basic events and also appear in the MCS.

Table 19-88-1 Common Cause Failure Events (sheet 1 of 3)

CCF Group Identifier	Component	Failure Mode	Basic Event Identifier	CCF group size	CCF Basic Event Identifier	Mean
CWSCF2CVOD052BD	CCW pump discharge line check valve 052B,052D	CCF of check valves to open	CWSCVOD052B CWSCVOD052D	2	CWSCF2CVOD052BD-ALL	5.6E-07
CWSCF2PCBDCWPBD	CCW pump B,D	CCF of pumps to start	CWSPCBDCWPB CWSPCBDCWPD	2	CWSCF2PCBDCWPBD-ALL	7.5E-05
CWSCF2PCYRCWPBD	CCW pump B,D	CCF of pumps to run 24 during the 24 hour mission time	CWSPCYRCWPB CWSPCYRCWPD	2	CWSCF2PCYRCWPBD-ALL	5.0E-06
CWSCF2RHPRHXBD	CS/RHR heat exchanger B,D	CCF of heat exchangers to PLUG during the 24 hour mission time	CWSRHPFCWHXB CWSRHPFCWHXD	2	CWSCF2RHPRHXBD-ALL	6.8E-08
SWSCF2CVOD502BD	ESW pump discharge line check valve 502B,502D	CCF of check valves to open	SWSCVOD502B SWSCVOD502D	2	SWSCF2CVOD502BD-ALL	5.6E-07
SWSCF2CVOD602BD	ESW pump cooling line check valve 602B,602D	CCF of check valves to open	SWSCVOD602B SWSCVOD602D	2	SWSCF2CVOD602BD-ALL	5.6E-07
SWSCF2PMBDSWPBD	ESW pump B,D	CCF of pumps to start	SWSPMBDSWPB SWSPMBDSWPD	2	SWSCF2PMBDSWPBD-ALL	1.4E-04
SWSCF2PMYRSWPAC	ESW pump A,C	CCF of pumps to run during the 24 hour mission time	SWSPMYRSWPA SWSPMYRSWPC	2	SWSCF2PMYRSWPAC-ALL	8.9E-06
SWSCF2PMYRSWPBD	ESW pump B,D	CCF of pumps to run during the 24 hour mission time	SWSPMYRSWPA SWSPMYRSWPC	2	SWSCF2PMYRSWPBD-ALL	8.9E-06

Table 19-88-1 Common Cause Failure Events (sheet 2 of 3)

CCF Group Identifier	Component	Failure Mode	Basic Event Identifier	CCF group size	CCF Basic Event Identifier	Mean
IECWSCF2PCYRA	CCW PUMP A,C (IE=A)	CCF of two normally running pumps during the reactor year. (Failure of pump A followed by failure of pump C)	Y-CWSPCYRCWPA N-CWSPCYRCWPC	2	IECWSCF2PCYRA-ALL	9.0E-04
IECWSCF2PCYRC	CCW PUMP A,C (IE=C)	CCF of two normally running pumps during the reactor year (Failure of pump C followed by failure of pump A)	Y-CWSPCYRCWPC N-CWSPCYRCWPA	2	IECWSCF2PCYRC-ALL	9.0E-04
IECWSCF2RHPRA	CCW HX A,C (IE=A)	CCF (PLUG) of heat exchangers of the two normally running trains during the reactor year (Failure in train A followed by failure in train C)	Y-CWSRHPFCWHXA N-CWSRHPFCWHXC	2	IECWSCF2RHPRA-ALL	1.2E-05
IECWSCF2RHPRC	CCW HX A,C (IE=C)	CCF of heat exchangers of the two normally running trains during the reactor year. (Failure in train C followed by failure in train A)	Y-CWSRHPFCWHXC N-CWSRHPFCWHXA	2	IECWSCF2RHPRC-ALL	1.2E-05
IESWSCF2PMYRA	ESW PUMP A,C (IE=A)	CCF of two normally running pumps during the reactor year (Failure of pump A followed by failure of pump C)	Y-SWSPMYRSWPA N-SWSPMYRSWPC	2	IESWSCF2PMYRA-ALL	1.6E-03
IESWSCF2PMYRC	ESW PUMP A,C (IE=C)	CCF of two normally running pumps during the reactor year (Failure of pump C followed by failure of pump A)	Y-SWSPMYRSWPC N-SWSPMYRSWPA	2	IESWSCF2PMYRC-ALL	1.6E-03

Table 19-88-1 Common Cause Failure Events (sheet 3 of 3)

CCF Group Identifier	Component	Failure Mode	Basic Event Identifier	CCF group size	CCF Basic Event Identifier	Mean
NA	CCWS pump A,B,C,D	CCF of two normally running pumps during the reactor year that also causes CCF to the two standby pumps to run	Y-CWSPCYRCWPA CWSPCYRCWPB N-CWSPCYRCWPC CWSPCYRCWPD	4	IE-CWSCF4PCYR-FF	2.5E-06
NA	CCWS Heat Exchanger	CCF in heat exchangers of two normally running trains during the reactor year that also causes CCF to that of the two standby trains	Y-CWSRHPFCWHXA CWSRHPFCWHXB N-CWSRHPFCWHXC CWSRHPFCWHXD	4	IE-CWSCF4RHPF-FF	1.3E-05
NA	ESWS pump A,B,C,D	CCF of two normally running pumps and two standby pumps to run during the 24 hour mission time	Y-SWSPMYRSWPA SWSPMYRSWPB N-SWSPMYRSWPC SWSPMYRSWPD	4	IE-SWSCF4PMYR-FF	4.4E-06
NA	CCWS pump A,B,C,D	CCF of two normally running pumps and two standby pumps during the 24 hour mission time	CWSPCYRCWPA CWSPCYRCWPB CWSPCYRCWPC CWSPCYRCWPD	4	CWSCF4PCYR-FF	6.7E-09
NA	CCWS Heat Exchanger	CCF in heat exchangers of two normally running trains and two standby trains during the 24 hour mission time	CWSRHPFCWHXA CWSRHPFCWHXB CWSRHPFCWHXC CWSRHPFCWHXD	4	CWSCF4RHPR-FF	3.6E-08
NA	ESWS pump A,B,C,D	CCF of two normally running pumps and two standby pumps during the 24 hour mission time	SWSPMYRSWPA SWSPMYRSWPB SWSPMYRSWPC SWSPMYRSWPD	4	SWSCF4PMYR-FF	1.2E-08

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.40-610 REVISION 0

SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

APPLICATION SECTION: PRA

DATE OF RAI ISSUE: 7/29/2008

QUESTION NO. : 19-89

It is stated (DCD Chapter 19, page 19.1-16) that "Fault tree definition includes the development of dependency matrices that identify .... the dependencies between initiating events and systems." The staff could not find where the dependencies between the initiating events and the systems are discussed and documented. Please clarify.

ANSWER:

MHI agrees that no descriptions about the dependencies between the initiating events and the systems in a matrix format are presented in the current version of DCD as pointed by the staff. The discussions will be added in the next revised DCD as well as the PRA technical report. For your information, the identified dependencies between the initiating events and the systems include the followings.

- LOCA: Considering that injected water flows out from the pipe break, it is assumed that safety injection to the broken loop (i.e. injection to cold leg for large LOCA and medium LOCA, and DVI for small LOCA) is not credited.

- Reactor vessel rupture: Considering the vessel breach beyond the design basis for the safety injection, any safety injection is not credited.

- Loss of feedwater flow: Main feedwater is not credited.

- SGTR, Steam line break, Feed line break: Secondary loop cooling by the broken SG is not credited.

- Total/partial loss of component cooling water: The system dependent on the function of the failed CCWS and ESWS is not credited. In case of total loss of component cooling water, it is assumed that RCP seal LOCA occurs since the integrity of reactor coolant pump seal cannot be maintained.

- Loss of offsite power: The systems energized from the non-safety power supply (e.g. main feedwater system, etc.) are not credited until the power supply is recovered. In case of loss of all ac power, it is assumed that RCP seal LOCA occurs since the integrity of reactor coolant pump seal cannot be maintained.

- Loss of class 1E 120V ac bus/loss of class 1E 125V dc bus: The systems energized from the ac/dc bus are not credited.

- Large LOCA: Some backup operations (e.g. operation of alternate core cooling, setup of AAC gas turbine generator, etc.) are not credited considering very limited time available for operation.

Impact on DCD

DCD will be revised to address the information discussed for this RAI.

Impact on COLA

This RAI and its response will impact the COLA, which refers the DCD.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.40-610 REVISION 0

SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

APPLICATION SECTION: PRA

DATE OF RAI ISSUE: 7/29/2008

QUESTION NO. : 19-90

The event trees for internal events and power operation are reported in Figure 19.1-1 of the DCD (19 sheets). The staff notes that there is no description of the event tree top events and that the success criteria for each top event are not stated in the DCD. Although this information is provided in the PRA document, a summary must be included in the DCD per RG 1.206 (Appendix C.I.19-A) guidance. For example, a table could be added that includes a brief description and the success criteria for each top event identifier.

ANSWER:

This question will be answered later, within 60 days after RAI issue date.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.40-610 REVISION 0**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-91

There is an apparent discrepancy between the success criteria for prevention of core damage (PRA report page 3-5) and the large LOCA (LLOCA) event tree (page 3-111). On page 3-5 of the PRA document, it is stated: "Heat removal from containment: The combination of CS/RHR (CV Spray injection) (CSA) and CS/RHR (Heat Removal) (CXC) or Alternate CV Cooling (FNA) is necessary." However, the LLOCA event tree shows CS/RHR (Heat Removal) alone, top event CXC, as providing adequate heat removal from the containment (e.g., accident sequence #4). Please clarify.

ANSWER:

Qualitatively, accident sequence #4 is success by the combination of CS/RHR (Alternate core cooling) and CS/RHR (Heat removal). The result of thermal-hydraulic analyses is shown in Table 5A2.3-1 in the PRA report. Therefore, Event tree is correct.

On the other hand, it is assumed that CS/RHR (Alternate core cooling) is not effective at Large LOCA, because there is not enough time to switchover. And so, quantitatively, it is conservatively evaluated that failure probability of CS/RHR (Alternate core cooling) is 1.0 for all sequences at Large LOCA. Description of success criteria is based on this quantitative evaluation.

Above discussions will be incorporated into the revised technical report [1).]

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.40-610 REVISION 0**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-92

On page 3-3 of the PRA document it is stated that the "Alternate core injection" function, top event CRC, is not credited for large LOCAs. However, Section 3.2.1.3.2 "Success Paths for Prevention of Core Damage," Table 3.2.1.2-1, the LLOCA event tree and Table 6A.3-2 "Success Criteria (Alternative Core Cooling) indicate that credit is taken for the "Alternate core injection" function (top event CRC). Please clarify and revise, as necessary.

ANSWER:

As discussed in No. 19-91, CS/RHR (Alternate core cooling) is qualitatively effective. But, quantitatively, it is conservatively evaluated that failure probability is 1.0 at Large LOCA.

Above discussions will be incorporated into the revised technical report [1].

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.40-610 REVISION 0**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-93

It is stated in Chapter 19 of the DCD (page 19.1-19) that the mitigating systems and operator actions in accident sequences are determined as given in Tables 19.1-10 and 19.1-11. The staff notices that some mitigating systems and operator actions are not included in these two tables, such as the Alternate Containment Cooling (CSR). Please explain and revise accordingly, as necessary.

ANSWER:

MHI understands NRC's request. Complete results of success criteria will be prepared as Table 19.1-12 discussed in RAI-#40 question 19-98. Also, Table 19.1-10 and Table 19.1-11 will be revised to meet with Table 19.1-12 in the next revision of the DCD..

Impact on DCD

DCD will be revised to address the information discussed for this RAI.

Impact on COLA

This RAI and its response will impact the COLA, which refers the DCD.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.40-610 REVISION 0

SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

APPLICATION SECTION: PRA

DATE OF RAI ISSUE: 7/29/2008

QUESTION NO. : 19-94

The definition and criteria of core damage are stated on page 19.1-19 of the DCD. It is stated that the measure (plant parameter) used for core damage is the "core peak-node temperature" and the acceptance criteria are (1) 2,200 degrees F predicted by a code with detailed core modeling, and (2) 1,400 degrees F predicted by a code with simplified modeling of the core. This statement implies that codes with detailed as well as simplified core modeling were used for thermal-hydraulic calculations to determine whether an accident sequence leads to core damage and the success criteria for the mitigating systems and operator actions. Please verify this statement and state in the DCD the names of the codes that were used and for what accident sequences a code with detailed core modeling was used. Also, please clarify the reference to peak clad temperature (PCT) while the measure selected is the "core peak-node temperature" instead.

ANSWER:

The code with detailed as well as simplified core modeling was used for thermal-hydraulic calculations to determine the success criteria for the mitigating systems and operator actions.

The code with detailed core modeling is "WCOBRA/TRAC(M1.0)" which is described in the DCD Chapter 15. This code is used for determining the success criteria for large pipe break LOCA. The result of the analysis is described in the PRA report Chap.5 attachment A. The code with simplified core modeling is "MAAP 4.0.6," which is used for determining success criteria for other events. The names of the analysis codes and corresponding accident sequences will be included in the DCD and the PRA report in the next time

The plant parameter used for core damage is the PCT, not the core peak node temperature. Description in the DCD is not correct. This editorial error will be corrected in the revised DCD and technical report 1).

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

<u>Impact on DCD</u>

DCD will be revised to address the information discussed for this RAI.

<u>Impact on COLA</u>

This RAI and its response will impact the COLA, which refers the DCD.

<u>Impact on PRA</u>

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.40-610 REVISION 0

SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

APPLICATION SECTION: PRA

DATE OF RAI ISSUE: 7/29/2008

QUESTION NO. : 19-95

The following statement is made in Chapter 19 of the DCD (page 19.1-19): "Twenty-four hours was selected as an allowable mission time for the sequences. If a stable plant condition cannot be achieved within 24 hours for a specific sequence, additional evaluation of the sequence is performed to determine an appropriate PDS, to extend the mission time, and/or to model additional system recovery." Please clarify this statement which implies that the mission time for some sequences is not 24 hours. Are there any sequences in the US-APWR PRA for which a stable plant condition cannot be achieved within 24 hours? If this is correct, please discuss such sequences and what additional systems or operator actions are needed at some time beyond the 24-hour period.

ANSWER:

This statement in the DCD represents a general discussion on mission time. There are no sequences for which the mission time is over 24 hours in the PRA of the US-APWR.

Associated information is provided in Attachment A and B.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

Attachment A for Question 19-95

In the PRA technical report 1), there is a subsection in which mission time is confused as margin for time that is used for calculating recovery probabilities with mean time to repair. This will be corrected in the next revision of PRA technical report as described in this attachment.

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

Attachment B for Question 19-95

In addition to the inconsistency as described in Attachment A, the mission time for fault tree identifier RSA is incorrect. In Table 6A.15.5-2 of the PRA technical report, the mission time should be modified from 100 hours to 24 hours. This will also be corrected in the next revision of PRA technical report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.40-610 REVISION 0**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **SRP**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-96

The following statement is made in Chapter 19 of the DCD (page 19.1-20): "MAAP 4.0.6 code as well as analysis results described in Chapter 15 are used to determine success criteria." However, all "typical" results of thermal/hydraulic (T/H) analysis shown in Table 19.1-12 (page 19.1-140), were obtained by using the MAAP code. Please explain what success criteria were determined by using Chapter 15 analysis results. Also, please discuss how the reliability of the MAAP 4.0.6 code results was verified so that these results can be used to determine PRA success criteria (e.g., benchmarking with a more sophisticated T/H code).

ANSWER:

Analysis code that used for determine success criteria:

In order to determine the success criteria for the mitigating systems and operator, WCOBRA/TRAC (M1.0) and MAAP4.0.6 are used for large pipe break LOCA analysis and for other analyses, respectively. Please refer to the response to question No.19-97 for details of success criteria and their analyses.

Reliability of analysis code:

Success criteria are determined not only from the results of MAAP 4.0.6 analyses but also from the engineering judgment in consideration of uncertainties of analysis code. Therefore the success criteria obtained have a sufficient margin of safety and are considered to be conservative. Results of analyses in DCD Chap.15 are also used to determine success criteria, in order to consider the uncertainty of analysis codes. Please refer to the response to question No.19-97 for individual discussions on the success criteria.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.40-610 REVISION 0**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-97

Table 19.1-12 of Chapter 19 of the DCD (page 19.1-140) shows "typical" results of thermal/hydraulic (T/H) analysis obtained by using the MAAP code. The first column, labeled "Accident Sequence Designator," does not give much information about the accident sequence that is analyzed. For some cases, Table 19.1-12 provides only a partial list of the equipment that is operating. For other sequences, Table 19.1-12 lists as operating a larger set of mitigating equipment than the minimum set required according to the success criteria (e.g., the last case assumes operation of all accumulators and all emergency feedwater pumps). This table is included in Chapter 19 of the DCD without any accompanying discussion. Please provide a complete definition of the sequences used in each case, state the objective of each case, and explain how these "typical" results have been used to determine PRA success criteria. In the third column, labeled "Results," please include the estimated core peak-node temperature instead of just stating whether the core damage criterion was met or not.

ANSWER:

Detailed information on the success criteria analysis is described in Chapter 5 of the technical report [1)]. Table 19.1-12 of DCD will be modified to contain detailed information on the success criteria as shown in Attachment A. In this table, description of all cases of success criteria analyses and the following items are included.

- Accident sequence number
- Objective of the analysis

- Accident sequence description
- Results
- Insights from success criteria analysis

The equivalent information will be also included in Chapter 5 of the technical report in the next revision.

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

Impact on DCD

DCD will be revised to address the information discussed for this RAI.

Impact on COLA

This RAI and its response will impact the COLA, which refers the DCD.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.40-610 REVISION 0**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-98

The following statement is made about the PRA success criteria on page 19.1-21 of the DCD: "Final success criteria, shown in Table 19.1.13, are determined from the design, engineering judgment and thermal/hydraulic analysis results in a manner that allows a margin for the uncertainties that attribute models of the thermal/hydraulic analyses and grouping of initiating events." This statement appears to imply that Table 19.1.13 provides a compete list of PRA success criteria which is not the case. A complete list of PRA success criteria should be included in the DCD (per R.G. 1.206, Section 19.1.4.1.1 of Appendix A). Also, please discuss (with reference to Appendix 5A "Thermal/Hydraulic Analysis for Success Criteria" of the PRA report) what design and engineering judgment results were used in conjunction with thermal/hydraulic (T/H) analysis results to account for code and T/H modeling uncertainties and for the purpose of grouping initiating events. A summary discussion, including important assumptions made, should be included in the DCD.

ANSWER:

Table 19.1-13 in the DCD will be replaced by a table that describes more detailed and complete information on success criteria (ref. Attachment A). The title of this table is also modified from "Results of Thermal/Hydraulic Analysis and Success Criteria" to "List of Success" . Summary discussions on how to determine success criteria will be also added on page 19.1-21 of the DCD. Detailed discussions on success criteria are described as follows. Please refer to the response to RAI-#40 question 19-97 for the accident sequence number described below.

<u>Large pipe break LOCA (LLOCA)</u>

(#1-2 : Core injection function)

Two accumulators and two safety injection pumps are sufficient for core injection function from the accident sequence No.1.1. Because MAAP dose not provide appropriate solutions for initial phase of large pipe break LOCA where momentum equation plays an important role, COBRA/TRAC is used for determination of success criteria instead. The success criteria for core injection function are determined from these results.

(#1 : Decay heat removal and containment heat removal function)

One CS/RHR pump and one heat exchange are sufficient from the sequence No.2.1. The success criteria for heat removal function are determined from these results.

(#2 : Decay heat removal and containment heat removal function)

From subsection 14.3.3.5 of the PRA technical report[1)], two CCW pumps and two containment fan cooler units used as alternate containment cooling are sufficient for decay heat removal and containment heat removal function. Although this analysis evaluates accident progression for hot leg 8 inches break leading to core damage and it is not the most appropriate analysis for determining success criteria, they are judged to be the same conditions as in the analysis because this function is required in enough time after onset of the accident when the heat transfer to containment depends roughly on decay heat which is common to almost all accident sequences.

<u>Medium pipe break LOCA (MLOCA)</u>

(#1-3 : Core injection function)

Two accumulators and one safety injection pump are sufficient for core injection function from the accident sequence No.1.1 and No.1.2. Furthermore, there is no need to heat removal from SGs according to the accident sequence No.1.3. The success criteria for core injection function are determined from these results.

(#1-2, 5 : Decay heat removal and containment heat removal function)

The discussions in LLOCA are applicable to MLOCA.

(#4-5 : Core injection function)

Two accumulators and one safety injection pump are sufficient for core injection function from the accident sequence No.1.1 and No.1.2. From the accident sequence No.1.5, four accumulators and one CS/RHR pump with four EFW pumps, four SGs, and three MSRVs are sufficient for core injection function. From these analysis results, the success criteria are judged to be two accumulators and one CS/RHR pump with three EFW pumps, three SGs, and three MSRVs. Accident sequences without accumulators do not lead to core damage as shown in the No.1.1 and No.1.2, which suggests that the number of accumulators has little impact on the accident progression. This is the reason why the number of accumulators is determined to be two. Furthermore, three EFW pumps and three SGs are adopted as success criteria instead of four pumps and four SGs considering the engineering judgment that there is little difference in accident progression between three and four EFW pumps to SGs.

From the accident sequence No.1.4, the margin for time to alternate core cooling for core injection is sufficient. This conclusion is common to all accident sequences.

(#3- 4 : Decay heat removal and containment heat removal function)

From the accident sequence No.2.3, one CS/RHR pump with four EFW pumps, four SGs, and four MSRVs are sufficient for decay heat removal and containment heat removal function. From these analysis results, the success criteria are judged to be one CS/RHR pump with three EFW pumps, three SGs, and three MSRVs. Because the accident sequence No.1.5 shows that whether three or four MSRVs open has little impact on the accident progression, success criteria are determined to be three MSRVs. Furthermore, three EFW pumps and three SGs are adopted as success criteria instead of four pumps and four SGs considering the engineering judgment that there is little difference in accident progression between three and four EFW pumps to SGs.

From the accident sequence No.2.2, the margin for time to alternate core cooling for heat removal is sufficient. This conclusion is common to all accident sequences.

Small pipe break LOCA (SLOCA)

(#1-7 : Reactor shutdown function)

Success criteria for reactor shutdown function are judged to be 66 out of 69 control rods, and two RPS or one DAS. The criteria for the control rods are determined from the following judgment.

The shutdown margin is determined from the assumption that the control rods are initially located at the insertion limit position and that following the trip signal all the control rods are inserted in the core except for the most reactive one which is conservatively assumed to remain at the top of the core. The required shutdown margin is 1.6% as shown in Table 4.3-2 of the DCD Chapter 4. On the other hand, maximum ejected and dropped worth used for analyses are at most 0.8% and 0.25%, respectively, as shown in Table 15.4.8-2 in the DCD subsection 15.4.8 and in the DCD subsection 15.4.3.3.1.2. It is estimated that the core does not become critical even on the conservative assumption that the second and the third most reactive rods are equivalent to the most reactive one and that these three rods fail to be inserted in the core at the same time. Furthermore, possibility of criticality is low even if more than three reactive rods fail to be inserted in the core. There is very low probability of such a failure. This is the reason why reactor shutdown is judged to succeed even when three rods fail to be inserted in the core.

(#1-3 : Core injection function)

One safety injection pump without EFWS is sufficient for core injection function from the accident sequence No.1.3. For smaller brake size LOCA, however, heat removal from SGs is necessary. From the accident sequence No.3.1, one EFW pump and one SG are sufficient for decay heat removal for transient sequences, which is applicable to SLOCA. The success criteria for core injection function are conservatively judged to be one safety injection pump, and two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened.

(#4-5 : Core injection function)

From the accident sequence No.1.5, four accumulators and one CS/RHR pump with four EFW pumps, four SGs, and three MSRVs are sufficient for core injection function. From these analysis results, the success criteria are judged to be one accumulator, and one CS/RHR pump with three EFW pumps, three SGs, and three MSRVs. MLOCA sequences without accumulators do not lead to core damage as shown in the No.1.1 and No.1.2, which suggests that the number of accumulators has little impact on smaller pipe break LOCA sequences. Here one accumulator is expected in the success criteria through the engineering judgment. Furthermore, three EFW pumps and three SGs are adopted as success criteria instead of four pumps and four SGs considering the engineering judgment that there is little difference in accident progression between three and four EFW pumps to SGs.

(#6-7 : Core injection function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for core injection function in transient sequences. The success criteria for core injection function are judged to be determined from these results because RCS pressure in this sequence is lower than in transient sequences and more flow rate of coolant injection is expected.

From the accident sequence No.3.3, the margin for time to feed and bleed is sufficient. This conclusion is common to all accident sequences.

(#1-7 : Decay heat removal and containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to SLOCA.

<u>Very small pipe break LOCA (VSLOCA)</u>

(#1-7 : Reactor shutdown function)

The discussions in SLOCA are applicable to VSLOCA.

(#1-3 : Core injection function)

Success criteria for core injection function are judged to be one safety injection pump or one CHP, and two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened. One CHP is capable of maintaining the RCS inventory for 3/8 inches break as described in subsection 9.3.4.2.7.4 of the DCD, which roughly supports the judgment for the success criteria. One safety injection pump is also appropriate as success criteria for this function because more flow rate is expected than in CHP. The number of EFW pumps and SGs is conservatively determined from the analysis results of the accident sequence No.3.1 which indicate that one EFW pump and one SG are sufficient for decay heat removal for transient sequences.

(#4-5 : Core injection function)

From the accident sequence No.1.5, four accumulators and one CS/RHR pump with four EFW pumps, four SGs, and three MSRVs are sufficient for core injection function. From these analysis results, the success criteria are judged to be one accumulator, and one CS/RHR pump with three EFW pumps, three SGs, and three MSRVs. MLOCA sequences without accumulators do not lead to core damage as shown in the No.1.1 and No.1.2, which suggests that the number of accumulators has little impact on smaller pipe break LOCA sequences. Here one accumulator is expected in the success criteria through the engineering judgment. Furthermore, three EFW pumps and three SGs are adopted as success criteria instead of four pumps and four SGs considering the engineering judgment that there is little difference in accident progression between three and four EFW pumps to SGs.

(#6-7 : Core injection function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for core injection function in transient sequences. The success criteria for core injection function are judged to be determined from these results because RCS pressure in this sequence is lower than in transient sequences and more flow rate of coolant injection is expected.

(#1-7 : Decay heat removal and containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to VSLOCA.

Steam generator tube rupture (SGTR)

(#1-6 : Reactor shutdown function)

The discussions in SLOCA are applicable to SGTR.

(#1 : Core injection function & Decay heat removal and containment heat removal function)

Success criteria for core injection and heat removal function are conservatively judged to be two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened. The analysis results of the accident sequence No.3.1, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment.

(#2 : Core injection function)

Success criteria for core injection are judged to one safety injection pump, RCS depressurization by secondary cooling and by one SDV, and injection control. The success criteria for RCS depressurization by secondary cooling are one EFW pump, one SG, and one MSRV or one EFW, one SG with tie-line opened, and one MSRV. Those success criteria are adopted because there is enough time before core damage occurs in SGTR.

(#2 : Decay heat removal and containment heat removal function)

Even in the sequences in which EFW is not effective such as LLOCA, one CS/RHR pump and one heat exchanger is sufficient for heat removal as shown in the accident sequences No.2.1 and 2.3. These results suggest that one CS/RHR pump and one heat exchanger is sufficient for heat removal in enough time after onset of SGTR. Therefore success criteria for heat removal are judged to be one CS/RHR pump and one heat exchanger.

(#3-6 : Core injection function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for core injection function in transient sequences. The success criteria for core injection function are judged to be determined from these results because RCS pressure is lower than in transient sequences and more flow rate of coolant injection is expected.

If RCS depressurization by secondary cooling and by SDV, and injection control are added to the above success criteria, they are also assured obviously.

(#3-6 : Decay heat removal and containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to SGTR.

<u>Steam line break downstream MSIV (SLBO)</u>

(#1-3 : Reactor shutdown function)

The discussions in SLOCA are applicable to SLBO.

(#1 : Decay heat removal function)

Success criteria for decay heat removal function are conservatively judged to be two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened, and three MSIVs. The analysis results of the accident sequence No.3.1, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment.

(#2-3 : Decay heat removal function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for decay heat removal function in transient sequences. The success criteria for decay heat removal function are judged to be the same conditions as in the analysis because the accident progression of SLBO is similar to that of loss of feedwater.

(#2-3 : Containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to SLBO.

<u>Steam line break upstream MSIV (SLBI)</u>

(#1-3 : Reactor shutdown function)

The discussions in SLOCA are applicable to SLBI.

(#1 : Decay heat removal function)

Success criteria for decay heat removal function are conservatively judged to be two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened, and three intact loop MSIVs or one broken loop main steam check valve. The analysis results of the accident sequence No.3.1, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment.

(#2-3 : Decay heat removal function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for decay heat removal function in transient sequences. The success criteria for decay heat removal function are judged to be the same conditions as in the analysis because the accident progression of SLBI is similar to that of loss of feedwater.

(#2-3 : Containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to SLBI.

Feedwater line break (FWLB)

(#1-3 : Reactor shutdown function)

The discussions in SLOCA are applicable to FWLB.

(#1 : Decay heat removal function)

Success criteria for decay heat removal function are conservatively judged to be two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened, and three intact loop MSIVs or one broken loop main steam check valve. The analysis results of the accident sequence No.3.1, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment.

(#2-3 : Decay heat removal function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for decay heat removal function in transient sequences. The success criteria for decay heat removal function are judged to be the same conditions as in the analysis because the accident progression of FWLB is similar to that of loss of feedwater.

(#2-3 : Containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to FWLB.

General transient (TRANS)

(#1-4 : Reactor shutdown function)

The discussions in SLOCA are applicable to TRANS.

(#1 : Decay heat removal function)

Success criteria for decay heat removal function are conservatively judged to be two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened. The analysis results of the accident sequence No.3.1, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment.

(#2 : Decay heat removal function)

Success criteria for decay heat removal function are conservatively judged to be two MFW pumps and two SGs. The analysis results of the accident sequence No.3.1, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment. The difference between EFW and MFW is less significant in the discussion of success criteria.

(#3-4 : Decay heat removal function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for decay heat removal function in transient sequences. The success criteria for decay heat removal function are judged to be the same conditions as in the analysis because the accident progression of TRANS is similar to that of loss of feedwater.

(#3-4 : Containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to TRANS.

Loss of feedwater flow (LOFF)

(#1-3 : Reactor shutdown function)

The discussions in SLOCA are applicable to LOFF.

(#1 : Decay heat removal function)

Success criteria for decay heat removal function are conservatively judged to be two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened. The analysis results of the accident sequence No.3.1, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment.

(#2-3 : Decay heat removal function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for decay heat removal function in transient sequences. The success criteria for decay heat removal function are determined from these results.

(#2-3 : Containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to LOFF.

Loss of component cooling water (LOCCW)

(#1 : Reactor shutdown function)

The discussions in SLOCA are applicable to LOCCW.

(#1 : Core injection function - seal injection)

Success criteria for seal injection judged to be one CHP, and one fire protection water supply pump or one non-essential chilled water pump. One CHP has sufficient capacity for seal water supply because seal water flow rate of CHP is low in comparison with normal charging flow rate as shown in Table 9.3.4-2 of the DCD Chapter 9. Furthermore, one fire protection water supply pump or one non-essential chilled water pump has sufficient capacity of cooling one CHP. This is the basis for the success criteria.

(#1 : Decay heat removal function)

Success criteria for decay heat removal function are judged to be two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened. The analysis results of the accident sequence No.3.2, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment. The accident sequence No.3.2 is referred because reactor shutdown occurs almost at the same time as the onset of LOCCW.

<u>Partial loss of component cooling water (PLOCW)</u>

(#1-4 : Reactor shutdown function)

The discussions in SLOCA are applicable to PLOCW.

(#1 : Decay heat removal and containment heat removal function)

When neither stuck open safety valve LOCA nor RCP seal LOCA occurs, success criteria for heat removal function are judged to be two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened. The analysis results of the accident sequence No.3.2, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment. The accident sequence No.3.2 is referred because reactor shutdown occurs almost at the same time as the onset of POCW.

(#2-3 : Core injection function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for core injection function in transient sequences. When stuck open safety valve LOCA or RCP seal LOCA occurs, success criteria for core injection function are judged to be determined from these results because RCS pressure in this sequence is lower than in transient sequences and more flow rate of coolant injection is expected.

(#4 : Core injection function)

The discussions in SLOCA are applicable to PLOCW.

(#2-4 : Decay heat removal and containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to PLOCW.

If two EFW pumps and two SGs or one EFW pump and two SGs are added to the above success criteria, they are also assured obviously.

Loss of offsite power (LOOP)

(#1-4 : Reactor shutdown function)

The discussions in SLOCA are applicable to LOOP.

(#1 : Decay heat removal and containment heat removal function)

Success criteria for heat removal function are judged to be two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened. The analysis results of the accident sequence No.3.2, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment.

(#2-3 : Core injection function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for core injection function in transient sequences. The success criteria for core injection function are judged to be determined from these results because RCS pressure in this sequence is lower than in transient sequences and more flow rate of coolant injection is expected.

(#4 : Core injection function)

The discussions in SLOCA are applicable to LOOP.

(#2-4 : Decay heat removal and containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to LOOP.

If two EFW pumps and two SGs or one EFW pump and two SGs are added to the success criteria based on the above discussions, they are also assured obviously.

Loss of vital AC bus (LOAC)

(#1 : Decay heat removal and containment heat removal function)

Success criteria for heat removal function are judged to be two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened. The analysis results of the accident sequence No.3.2, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment.

(#2 : Decay heat removal and containment heat removal function)

Success criteria for heat removal function are judged to be two MFW pumps and two SGs. The analysis results of the accident sequence No.3.2, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment. The difference between EFW and MFW is less significant in the discussion of success criteria.

(#3-4 : Core injection function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for core injection function in transient sequences. The success criteria for core injection function are judged to be the same conditions as in the analysis because the accident progression of LOAC is similar to that of loss of feedwater.

(#3-4 : Containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to LOAC.

Loss of vital DC bus (LODC)

(#1 : Decay heat removal and containment heat removal function)

Success criteria for heat removal function are judged to be two EFW pumps and two SGs or one EFW pump and two SGs with tie-line opened. The analysis results of the accident sequence No.3.2, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment.

(#2 : Decay heat removal and containment heat removal function)

Success criteria for heat removal function are judged to be two MFW pumps and two SGs. The analysis results of the accident sequence No.3.2, which indicate that one EFW pump and one SG are sufficient for decay heat removal, support this judgment. The difference between EFW and MFW is less significant in the discussion of success criteria.

(#3-4 : Core injection function)

From the accident sequence No.3.4, one safety injection pump and one SDV are sufficient for core injection function in transient sequences. The success criteria for core injection function are judged to be the same conditions as in the analysis because the accident progression of LODC is similar to that of loss of feedwater.

(#3-4 : Containment heat removal function)

The discussions in LLOCA and MLOCA are applicable to LODC.

Anticipated transient without scram (ATWS)

(#1-3 : Reactor shutdown function)

Success criteria for reactor shutdown function are combinations of 66 out of 69 control rods, support software, two RPS, and one DAS. The criteria for the control rods are determined according to the discussions in SLOCA.

(#3 : Preventing function)

Success criteria for preventing function are judged to be MTC within allowable range, four pressurizer safety valves, heat removal from SGs, and turbine trip. These criteria, especially for MTC, are determined from the following judgment.

It is assumed conservatively that the time rate during which MTC is favorable for prevention of RCS pressure increase is more than 95% of the total core life. Time rate during which MTC is unfavorable is 10% for non-turbine trip events and 1% for turbine trip events. Because the frequency of turbine trip events is higher than of non-turbine trip events, it is estimated that time rate during which MTC is unfavorable is about 3%. This is the basis of the above assumption.

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

<u>Impact on DCD</u>

DCD will be revised to address the information discussed for this RAI.

<u>Impact on COLA</u>

This RAI and its response will impact the COLA, which refers the DCD.

<u>Impact on PRA</u>

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

Attachment A for Question 19-98

Table 19.1-13 will be replaced by the following table that describes detailed success criteria.

Table 19.1-13 List of Success Criteria (Sheet 1 of 16)
Large Pipe Break LOCA (>8 inches) Event Success Criteria

	Core injection function			Decay heat removal & containment heat removal function		
	Accumulator system	High head injection system	CS/RHR (Alternate core cooling)	CS/RHR (Containment spray) and CS/RHR (Heat removal)	CS/RHR (Alternate core cooling) [(2)] and CS/RHR (Heat removal)	Alternate containment cooling
1	2/3 ACCs[(1)]	2/4 SIPs [(1)]	-	1/4 CS/RHR pump and heat exchanger	-	-
2	2/3 ACCs[(1)]	2/4 SIPs [(1)]	-	-	-	2/4 CCWPs and 2/4 Containment fan cooler units

Note(1): RCS cold leg pipe break is assumed for large pipe break LOCA. Accumulator injection via the broken line is unavailable, and high head injection via DVI lines is available.

Note(2): Require operator action to change line-up to low pressure injection mode from CS/RHR(Containment spray) mode. For large pipe break LOCA, this mitigation system is assumed to be unavailable because there is not enough time to operate before core damage.

Table 19.1-13 List of Success Criteria (Sheet 2 of 16)
Medium Pipe Break LOCA (2 – 8 inches) Event Success Criteria

	Core injection function				Decay heat removal & containment heat removal function		
	Accumulator system	High head injection system	CS/RHR (Alternate core cooling) (2)	RCS depressurization by secondary side cooling	CS/RHR (Containment spray) and CS/RHR (Heat removal)	CS/RHR (Alternate core cooling) (2) and CS/RHR (Heat removal)	Alternate containment cooling
1	2/3 ACCs(3)	1/3 SIP (1)	-	-	1/4 CS/RHR pump and heat exchanger	-	-
2	2/3 ACCs(3)	1/3 SIP (1)	-	-	-	-	2/4 CCWPs and 2/4 Containment fan cooler units
3	2/3 ACCs(3)	1/3 SIP (1)	-	3/4 SGs and 3/4 EFW pumps and 3/4 MSRVs opened	-	1/3 CS/RHR pump and heat exchanger (3)	-
4	2/3 ACCs(3)	-	1/3 CS/RHR pump(3)	3/4 SGs and 3/4 EFW pumps and 3/4 MSRVs opened	-	1/3 CS/RHR pump and heat exchanger (3)	-
5	2/3 ACCs(3)	-	1/3 CS/RHR pump(3)	3/4 SGs and 3/4 EFW pumps and 3/4 MSRVs opened	-	-	2/4 CCWPs and 2/4 Containment fan cooler units

Note(1): DVI pipe break is assumed for high head injection system. High head injection via the broken line is unavailable.
Note(2): Require operator action to change line-up to low pressure injection mode from CS/RHR(Containment spray) mode.
Note(3): RCS cold leg pipe break is assumed for alternate core cooling and accumulator injection. Alternate core cooling and accumulator injection via the broken line is unavailable.

Table 19.1-13 List of Success Criteria (Sheet 3 of 16)
Small Pipe Break LOCA (1/2 – 2 inches) Event Success Criteria

	Rector shutdown function	Core injection function						Decay heat removal & containment heat removal function		
	Reactor trip	Accumulator system	High head injection system	CS/RHR (Alternate core cooling) (2)	Heat removal via SGs	RCS depressurization by secondary side cooling	Safety depressurization valve	CS/RHR (Containment spray) and CS/RHR (Heat removal)	CS/RHR (Alternate core cooling) (2) and CS/RHR (Heat removal)	Alternate containment cooling
1	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/3 SIP (1)	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	1/4 CS/RHR pump and heat exchanger	-	-
2	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/3 SIP (1)	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	-	-	2/4 CCWPs and 2/4 Containment fan cooler units
3	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/3 SIP (1)	1/4 CS/RHR pump (3)	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	3/4 SGs and 3/4 EFW pumps and 3/4 MSRVs opened	-	-	1/4 CS/RHR pump and heat exchanger (3)	-

Table 19.1-13 List of Success Criteria (Sheet 3 of 16)
Small Pipe Break LOCA (1/2 – 2 inches) Event Success Criteria

	Rector shutdown function	Core injection function			Decay heat removal & containment heat removal function					
4	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	1/4 ACC (3)	-	1/4 CS/RHR pump (3)	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	3/4 SGs and 3/4 EFW pumps and 3/4 MSRVs opened	-	-	1/4 CS/RHR pump and heat exchanger (3)	-
5	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	1/4 ACC (3)	-	1/4 CS/RHR pump (3)	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	3/4 SGs and 3/4 EFW pumps and 3/4 MSRVs opened	-	-	-	2/4 CCWPs and 2/4 Containme nt fan cooler units
6	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/3 SIP (1)	-	-	-	1/2 SDV	1/4 CS/RHR pumps and heat exchangers	-	-
7	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/3 SIP (1)	-	-	-	1/2 SDV	-	-	2/4 CCWPs and 2/4 Containme nt fan cooler units

Note(1): DVI pipe break is assumed for high head injection. High head injection via the broken line is unavailable.
Note(2): Require operator action to change line-up to low pressure injection mode from CS/RHR(Containment spray) mode.

Note(3): Even if RCS cold leg pipe break is assumed for alternate core cooling and accumulator injection, alternate core cooling and accumulator injection via RCS cold leg pipe is available because of a little spilled water.

Table 19.1-13 List of Success Criteria (Sheet 4 of 16)
Very Small Pipe Break LOCA (<1/2 inches) Event Success Criteria

	Rector shutdown function	Core injection function			Decay heat removal & containment heat removal function					
	Reactor trip	Accumulator system	High head injection system OR Charging injection	CS/RHR (Alternate core cooling) (1)	Heat removal via SGs	RCS depressurization by secondary side cooling	Safety depressurization valve	CS/RHR (Containment spray) and CS/RHR (Heat removal)	CS/RHR (Alternate core cooling) (1) and CS/RHR (Heat removal)	Alternate containment cooling
1	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/4 SIP OR 1/2 CHP	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	1/4 CS/RHR pump and heat exchanger	-	-
2	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/4 SIP OR 1/2 CHP	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	-	-	2/4 CCWPs and 2/4 Containment fan cooler units

Table 19.1-13 List of Success Criteria (Sheet 4 of 16)
Very Small Pipe Break LOCA (<1/2 inches) Event Success Criteria

	Rector shutdown function	Core injection function			Decay heat removal & containment heat removal function					
3	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/4 SIP OR 1/2 CHP	1/4 CS/RHR pump	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	3/4 SGs and 3/4 EFW pumps and 3/4 MSRVs opened	-	-	1/4 CS/RHR pump and heat exchanger	-
4	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	1/4 ACC	-	1/4 CS/RHR pump	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	3/4 SGs and 3/4 EFW pumps and 3/4 MSRVs opened	-	-	1/4 CS/RHR pump and heat exchanger	-
5	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	1/4 ACC	-	1/4 CS/RHR pump	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	3/4 SGs and 3/4 EFWs pumps and 3/4 MSRV opened	-	-	-	2/4 CCWPs and 2/4 Containme nt fan cooler units

Table 19.1-13 List of Success Criteria (Sheet 4 of 16)
Very Small Pipe Break LOCA (<1/2 inches) Event Success Criteria

	Rector shutdown function	Core injection function				Decay heat removal & containment heat removal function				
6	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/4 SIP	-	-	-	1/2 SDV	1/4 CS/RHR pump and heat exchanger	-	-
7	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/4 SIP	-	-	-	1/2 SDV	-	-	2/4 CCWPs and 2/4 Containment fan cooler units

Note(1): Require operator action to change line-up to low pressure injection mode from CS/RHR(Containment spray) mode.

Table 19.1-13 List of Success Criteria (Sheet 5 of 16)
Steam Generator Tube Rupture Event Success Criteria

	Condition		Rector shutdown function	Core injection function			Decay heat removal & containment heat removal function			
	Isolation of faulted SG [1]	Heat removal via SGs	Reactor trip	High head injection system	Safety depressurization valve	RCS depressurization by secondary side cooling [2] and RCS depressurization by SDV [3] and Injection control [4]	Heat removal via SGs	CS/RHR (Containment spray) and CS/RHR (Heat removal)	CS/RHR (RHR operation) [5] and CS/RHR (Heat removal)	Alternate containment cooling
1	Succeeded	Succeeded	66/69 control rods and 2/4 RPSs OR 1/1 DAS	-	-	-	2/3 SGs and 2/3 EFW pumps OR 2/3 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	-
2	Failed	Succeeded	66/69 control rods and 2/4 RPSs OR 1/1 DAS	1/4 SIP	-	X	2/3 SGs and 2/3 EFW pumps OR 2/3 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	1/4 CS/RHR pump and heat exchanger	-

Table 19.1-13 List of Success Criteria (Sheet 5 of 16)
Steam Generator Tube Rupture Event Success Criteria

	Condition		Rector shutdown function	Core injection function			Decay heat removal & containment heat removal function			
3	Failed	Succeeded	66/69 control rods and 2/4 RPSs OR 1/1 DAS	1/4 SIP	1/2 SDV	X	2/3 SGs and 2/3 EFW pumps OR 2/3 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	1/4 CS/RHR pump and heat exchanger	-	-
4	Failed	Succeeded	66/69 control rods and 2/4 RPSs OR 1/1 DAS	1/4 SIP	1/2 SDV	X	2/3 SGs and 2/3 EFW pumps OR 2/3 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	2/4 CCWPs and 2/4 Containment fan cooler units
5	Succeeded	Failed	66/69 control rods and 2/4 RPSs OR 1/1 DAS	1/4 SIP	1/2 SDV	-	-	1/4 CS/RHR pump and heat exchanger	-	-
6	Succeeded	Failed	66/69 control rods and 2/4 RPSs OR 1/1 DAS	1/4 SIP	1/2 SDV	-	-	-	-	2/4 CCWPs and 2/4 Containment fan cooler units

Note(1): Closing the following valves for faulted SG isolation, EFW isolation valve and {(main steam relief valve or main steam relief valve block valve) and (MSIV or turbine bypass valve) and main steam safety valve}.
Note(2): 1/3 SG and 1/3 EFW pumps and 1/3 MSRV opened, OR 1/3 SG and 1/4 EFW pumps and isolation valves of pump discharge tie-line opened and 1/3 MSRV opened
Note(3): 1/2 SDV
Note(4): 1/2 CHP and Injection control

Note(5): Requires operator action to change line-up to RHR operation mode

Table 19.1-13 List of Success Criteria (Sheet 6 of 16)
Steam Line Break Downstream MSIV Event Success Criteria

	Rector shutdown function	Decay heat removal function				Containment heat removal function	
	Reactor trip	Heat removal via SGs	Main steam line isolation	High head injection system	Safety depressurization valve	CS/RHR (Containment spray) and CS/RHR (Heat removal)	Alternate containment cooling
1	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	3/4 MSIVs closed	-	-	-	-
2	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	-	1/4 SIP	1/2 SDV	1/4 CS/RHR pump and heat exchangers	-
3	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	-	1/4 SIP	1/2 SDV	-	2/4 CCWPs and 2/4 Containment fan cooler units

Table 19.1-13 List of Success Criteria (Sheet 7 of 16)
Steam Line Break Upstream MSIV Event Success Criteria

	Rector shutdown function	Decay heat removal function				Containment heat removal function	
	Reactor trip	Heat removal via SGs	Main steam line isolation	High head injection system	Safety depressurization valve	CS/RHR (Containment spray) and CS/RHR (Heat removal)	Alternate containment cooling
1	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	2/3 SGs and 2/3 EFW pumps OR 2/3 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	3/3 intact loop MSIVs closed OR 1/1 broken loop Main steam check valve closed	-	-	-	-
2	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	-	1/4 SIP	1/2 SDV	1/4 CS/RHR pump and heat exchanger	-
3	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	-	1/4 SIP	1/2 SDV	-	2/4 CCWPs and 2/4 Containment fan cooler units

Table 19.1-13 List of Success Criteria (Sheet 8 of 16)
Feedwater Line Break Event Success Criteria

	Rector shutdown function	Decay heat removal function				Containment heat removal function	
	Reactor trip	Heat removal via SGs	Main steam line isolation	High head injection system	Safety depressurization valve	CS/RHR (Containment spray) and CS/RHR (Heat removal)	Alternate containment cooling
1	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	2/3 SGs and 2/3 EFW pumps OR 2/3 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	3/3 intact loop MSIVs closed OR 1/1 broken loop Main steam check valve closed	-	-	-	-
2	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	-	1/4 SIP	1/2 SDV	1/4 CS/RHR pump and heat exchanger	-
3	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	-	1/4 SIP	1/2 SDV	-	2/4 CCWPs and 2/4 Containment fan cooler units

Table 19.1-13 List of Success Criteria (Sheet 9 of 16)
General Transient Event Success Criteria

	Rector shutdown function	Decay heat removal function			Containment heat removal function	
	Reactor trip	Feed and Bleed	Heat removal via SGs	Main feed water recovery	CS/RHR (Containment spray) and CS/RHR (Heat removal)	Alternate containment cooling
1	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	-
2	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	-	2/4 SGs and 1/4 MFW pump	-	-
3	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	1/4 SIP and 1/2 SDV	-	-	1/4 CS/RHR pump and heat exchanger	-
4	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	1/4 SIP and 1/2 SDV	-	-	-	2/4 CCWPs and 2/4 Containment fan cooler units

Table 19.1-13 List of Success Criteria (Sheet 10 of 16)
Loss of Feedwater Flow Event Success Criteria

	Rector shutdown function	Decay heat removal function		Containment heat removal function	
	Reactor trip	Heat removal via SGs	Feed and bleed	CS/RHR (Containment spray) and CS/RHR (Heat removal)	Alternate containment cooling
1	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	-
2	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/4 SIP and 1/2 SDV	1/4 CS/RHR pump and heat exchanger	-
3	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	1/4 SIP and 1/2 SDV	-	2/4 CCWPs and 2/4 Containment fan cooler units

Table 19.1-13 List of Success Criteria (Sheet 11 of 16)
Loss of Component Cooling Water Event Success Criteria

	Condition	Rector shutdown function	Core injection function	Decay heat removal function
	Stuck open safety valve LOCA(1)	Reactor trip	Alternate component cooling (Seal injection) (2)	Heat removal via SGs
1	Not occurred	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	1/2 CHP and 1/2 Fire protection water supply pump OR 1/1 Non-essential chilled water pump	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened

Note(1): Occurrence of stuck open safety valve LOCA during this initiating event is assumed to result in core damage.
Note(2): RCP seal LOCA is assumed to occur, when alternate component cooling fails.

Table 19.1-13 List of Success Criteria (Sheet 12 of 16)
Partial Loss of Component Cooling Water Event Success Criteria

	Condition	Rector shutdown function	Core injection function		Decay heat removal & containment heat removal function			
	Stuck open safety valve LOCA OR RCP seal LOCA (6)	Reactor trip	High head injection system(1) and Safety depressurization valve (2)	RCS depressurization by secondary side cooling(3) and Accumulator system(4) and CS/RHR (Alternate core cooling) (5)	Heat removal via SGs	CS/RHR (Containment spray) and CS/RHR (Heat removal)	CS/RHR (Alternate core cooling) and CS/RHR (Heat removal)	Alternate containment cooling
1	Not occurred	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	-
2	Occurred	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	X	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	1/4 CS/RHR pump and heat exchanger		

Table 19.1-13 List of Success Criteria (Sheet 12 of 16)
Partial Loss of Component Cooling Water Event Success Criteria

	Condition	Rector shutdown function	Core injection function		Decay heat removal & containment heat removal function			
3	Occurred	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	X	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	2/4 CCWPs and 2/4 Containment fan cooler units
4	Occurred	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	X	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	1/4 CS/RHR pump and heat exchanger	

Note(1): 1/4 SIP
Note(2): 1/2 SDV
Note(3): 3/4 SG and 3/4 EFW pumps and 3/4 MSRV opened
Note(4): 1/4 ACC
Note(5): 1/4 CS/RHR pumps
Note(6): RCP seal LOCA is assumed to occur when RCP seal cooling by the stand-by charging pump fails.

Table 19.1-13 List of Success Criteria (Sheet 13 of 16)
Loss of Offsite Power Event Success Criteria

	Condition	Rector shutdown function	Core injection function		Decay heat removal & containment heat removal function			
	Stuck open safety valve LOCA or RCP seal LOCA [(4)]	Reactor trip	Feed and Bleed	RCS depressurization by secondary side cooling [(1)] and Accumulator system [(2)] and CS/RHR (Alternate core cooling) [(3)]	Heat removal via SGs	CS/RHR (Containment spray) and CS/RHR (Heat removal)	CS/RHR (Alternate core cooling) and CS/RHR (Heat removal)	Alternate containment cooling
1	Not occurred	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	-
2	Occurred	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	1/4 SIP and 1/2 SDV	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	1/4 CS/RHR pump and heat exchanger		
3	Occurred	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	1/4 SIP and 1/2 SDV	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	2/4 CCWPs and 2/4 Containment fan cooler units

Table 19.1-13 List of Success Criteria (Sheet 13 of 16)
Loss of Offsite Power Event Success Criteria

	Condition	Rector shutdown function	Core injection function		Decay heat removal & containment heat removal function			
4	Occurred	2/4 RPSs and 66/69 control rods OR 1/1 DAS and 66/69 control rods	-	X	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	1/4 CS/RHR pump and heat exchanger	

Note(1): 3/4 SG and 3/4 EFW pumps and 3/4 MSRV opened
Note(2): 1/4 ACC
Note(3): 1/4 CS/RHR pumps
Note(4): RCP seal LOCA is assumed to occur when all CCW pumps fail to restart and alternate component cooling fails.

Table 19.1-13 List of Success Criteria (Sheet 14 of 16)
Loss of Vital AC Bus Event Success Criteria

	Core injection function	Decay heat removal & containment heat removal function			
	Feed and Bleed	Heat removal via SGs	Main feed water recovery	CS/RHR (Containment spray) and CS/RHR (Heat removal)	Alternate containment cooling
1	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pump and isolation valves of pump discharge tie-line opened	-	-	-
2	-	-	2/4s SG and 1/4 MFW pump	-	-
3	1/4 SIP and 1/2 SDV	-	-	1/4 CS/RHR pump and heat exchanger	-
4	1/4 SIP and 1/2 SDV	-	-	-	2/4 CCWPs and 2/4 Containment fan cooler units

Table 19.1-13 List of Success Criteria (Sheet 15 of 16)
Loss of Vital DC Bus Event Success Criteria

	Core injection function	Decay heat removal & containment heat removal function			
	Feed and Bleed	Heat removal via SGs	Main feed water recovery	CS/RHR (Containment spray) and CS/RHR (Heat removal)	Alternate containment cooling
1	-	2/4 SGs and 2/4 EFW pumps OR 2/4 SGs and 1/4 EFW pumps and isolation valves of pump discharge tie-line opened	-	-	-
2	-	-	2/4 SGs and 1/4 MFW pump	-	-
3	1/4 SIP and 1/2 SDV	-	-	1/4 CS/RHR pump and heat exchanger	-
4	1/4 SIP and 1/2 SDV	-	-	-	2/4 CCWPs and 2/4 Containment fan cooler units

Table 19.1-13 List of Success Criteria (Sheet 16 of 16)
Anticipated transient without scram Event Success Criteria

	Rector shutdown function				Preventing function
	Reactor trip (Control rod)	Reactor trip (Software CCF) (1)	Reactor trip (Excluding control rod and software CCF)	Diverse actuation system	Moderator temperature coefficient
1	66/69 control rods	X	2/4 RPSs	-	-
2	66/69 control rods	-	-	X	-
3	-	X	-	-	MTC within allowable range(95% of fuel cycle) and 4/4 pressurizer safety valves opened and Heat removal via SGs (4/4 SG and 4/4 EFW pumps) and turbine trip

Note(1): This software means "support software" which is common to all processors regardless of the application software installed in the CPU.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.40-610 REVISION 0**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **7/29/2008**

QUESTION NO. : 19-99

Several event trees (e.g., MLOCA, SLOCA and SGTR) assume that containment heat removal by the CS/RHR system is possible (e.g. top event CXC in the MLOCA event tree) given failure of containment spray injection by the CS/RHR system (e.g., top event CSA in the MLOCA event tree). Based on the definitions of top events CSA and CXC (Section 3.2.2.2 of the PRA report), it appears that the only difference between these two top events is that CXC requires operation of the CS/RHR heat exchangers while CSA does not. If this is correct, CXC cannot succeed when CSA fails. However, a branch is shown in the event tree indicating that CXC can succeed even when CSA fails. Please explain.

ANSWER:

Fault trees of CS/RHR are modeled as shown below.

1) CS/RHR (containment spray, CSA): failure of spray injection function is modeled as described in Figure 19-99.1.

2) CS/RHR (Alternate core cooling, CRD): failure of cold-leg injection function including failure of re-lineup operation is modeled as described in Figure 19-99.2.

3) CS/RHR (Heat removal, CXC): failure of cooling heat exchangers by opening MOV-114 (isolation valves of CCW) is modeled.

Because fault tree linking methods are applied for each sequence analysis, CXC is unable to function when CSA fails, e.g. CS/RHR pumps fails described in Figure 19-99.1, as you pointed out.

However, CXC sometimes succeeds even when CSA fails, e.g. MOV-9011 described in Figure 19-99.1 fails to open. Event trees represent possible branches for which dependent failure probabilities are quantified by fault tree linking methods considering these interactions.

The above discussions will be incorporated into the revised technical report [1].

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

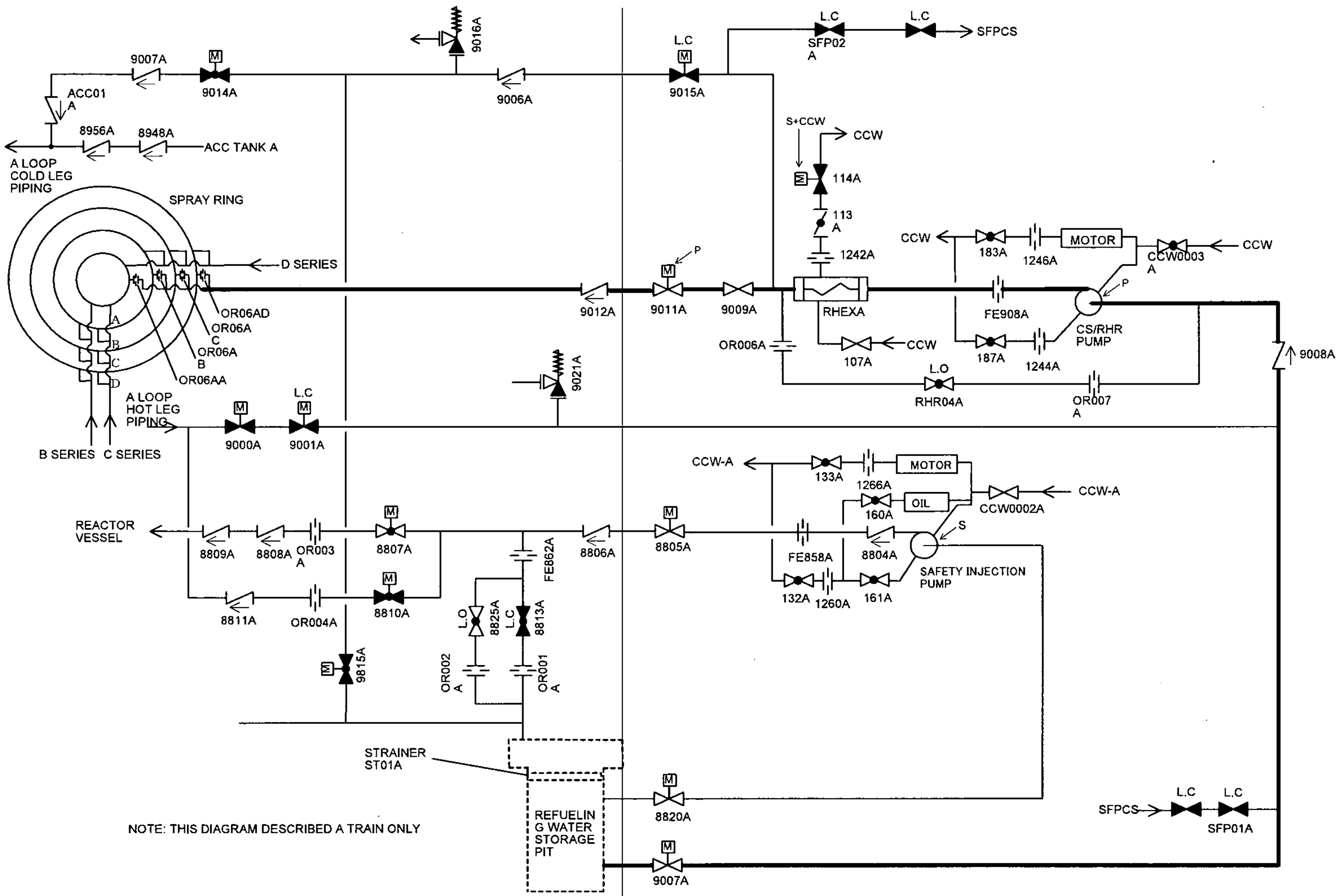

Figure 19-99.1 Containment spray function

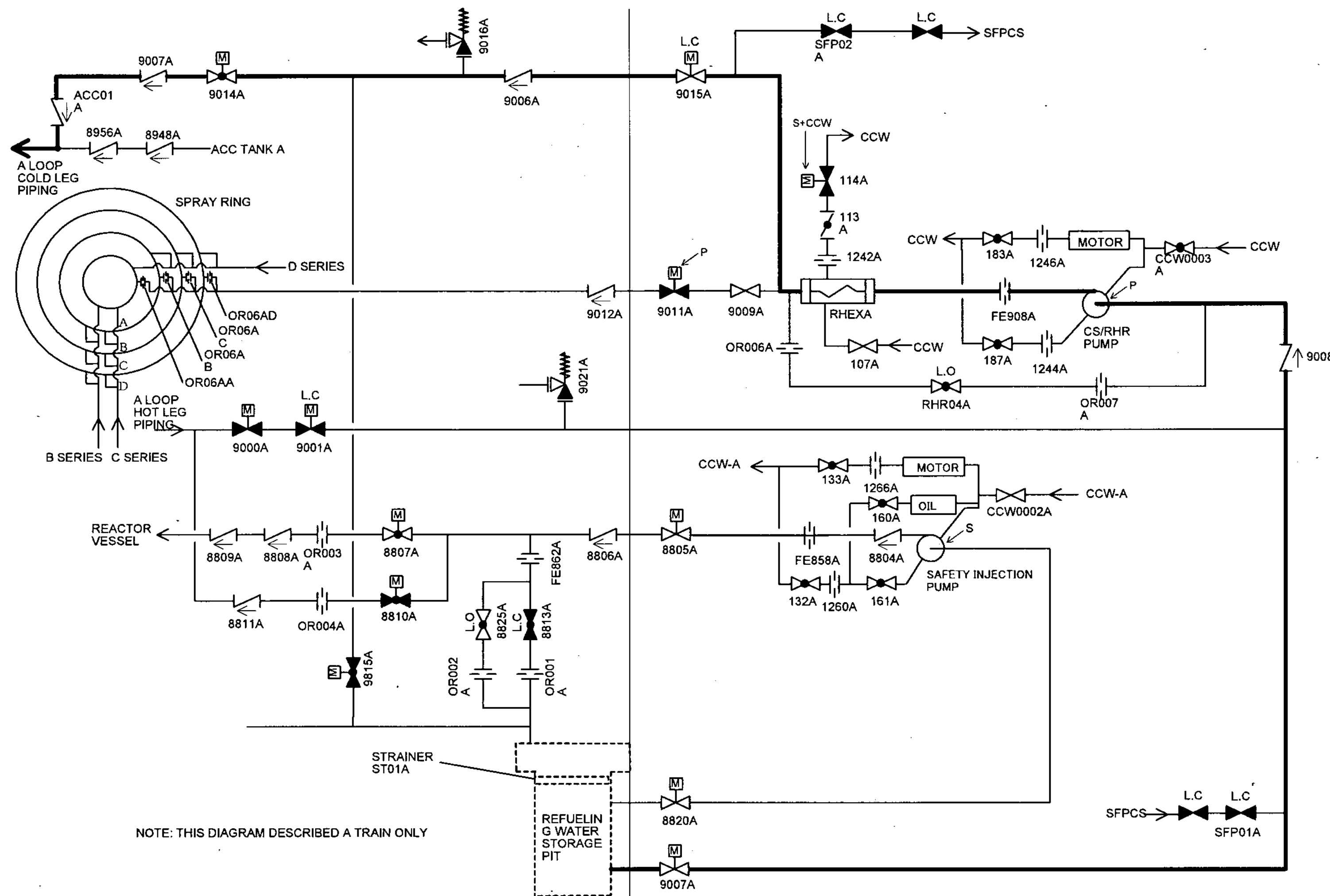

Figure 19-99.2 Alternate core cooling function

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/28/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.40-610 REVISION 0

SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

APPLICATION SECTION: PRA

DATE OF RAI ISSUE: 7/29/2008

QUESTION NO. : 19-100

It is stated in the PRA report (page 3-12) that top event SRA "Secondary side cooling to depressurize the RCS" is effective when "CS/RHR (Spray injection) System is not available, this measure depresses RCS pressure and enables to actuate CS/RHR (alternate injection System) and CS/RHR (heat removal) System." It is not clear how CS/RHR (heat removal) can be successful given CS/RHR (spray injection) is unavailable. Also, terminology used in Table 3.2.3.3-1 "Small LOCA Event Success Criteria" is inconsistent and confusing. For example, while the definition of "RCS depressurization by secondary side cooling" includes both operation of EFW and opening of MSRVs, only the success criteria for the MSRVs are listed (with a note stating that 2 EFW pumps are also needed while the previous column shows that 3 EFW pumps are needed). In addition, Table 3.2.3.2-1 (page 3-81) shows two fault tree identifiers (RSS-CSS-HR and RSS-RHR-HRSL) associated with SLOCA event tree top event CXB. This appears to be conflicting with Table 6A.3-2 "Success Criteria (Containment Spray)" which shows CXB associated only with fault tree identifier RSSRHR- HRSL. Please clarify.

ANSWER:

The basic evaluation of mitigating systems is as follows.

The high head injection system (HHIS) is used for coolant injection into the core when LOCA occurs. If HHIS fails, then alternate core cooling is used for coolant injection. The name of the operator actions is "CS/RHR (Alternate core cooling)." In an accident sequence whose RCS pressure remains at relatively high level, e.g. in SLOCA, however, the CS/RHR (Alternate core cooling) cannot function by itself due to low shutoff pressure of CS/RHR pumps. CS/RHR (Alternate core cooling) is available when RCS depressurization by secondary side cooling succeeds.

The above discussions will be incorporated into the revised PRA technical report [1].

Inconsistency in the terminology used in success criteria and event tree top events will be corrected. Amended tables and figures of chapter 3 of the PRA technical report is provided in attachment.

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

Impact on DCD

There is no impact on DCD from this RAI as the response contains only additional information.

Impact on COLA

There is no impact on COLA from this RAI as the response contains only additional information.

Impact on PRA

There is no impact on PRA from this RAI and the response as this additional information does not cause any changes to the PRA.

Attachment

Amended tables and figures used in chapter 3 of the PRA technical report revision 0 is provided in this attachment. This attachment contains the following tables and figures of each event tree modeled in the PRA.

- Table describing description of event heading and branch
- Table describing event success criteria
- Figure of event tree

Node ID is assigned to each node of the event trees. For each node that does not have numbering in the event tree figures, node ID "1" is assigned. Fault tree or basic events that are considered for each node is shown in tables describing descriptions of event heading and branch.

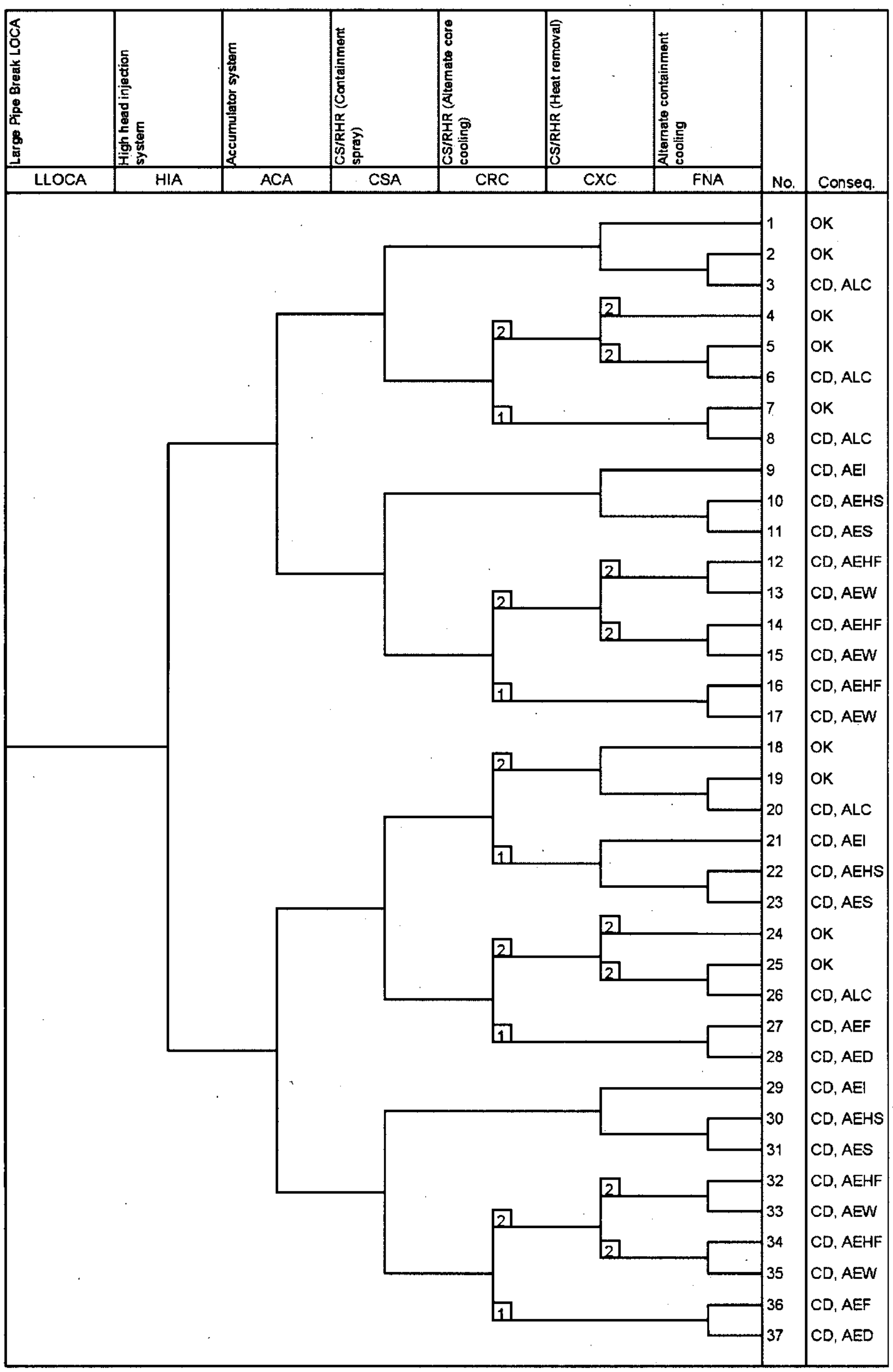

Figure 3.2.1-1 Large Pipe Break LOCA Event Tree

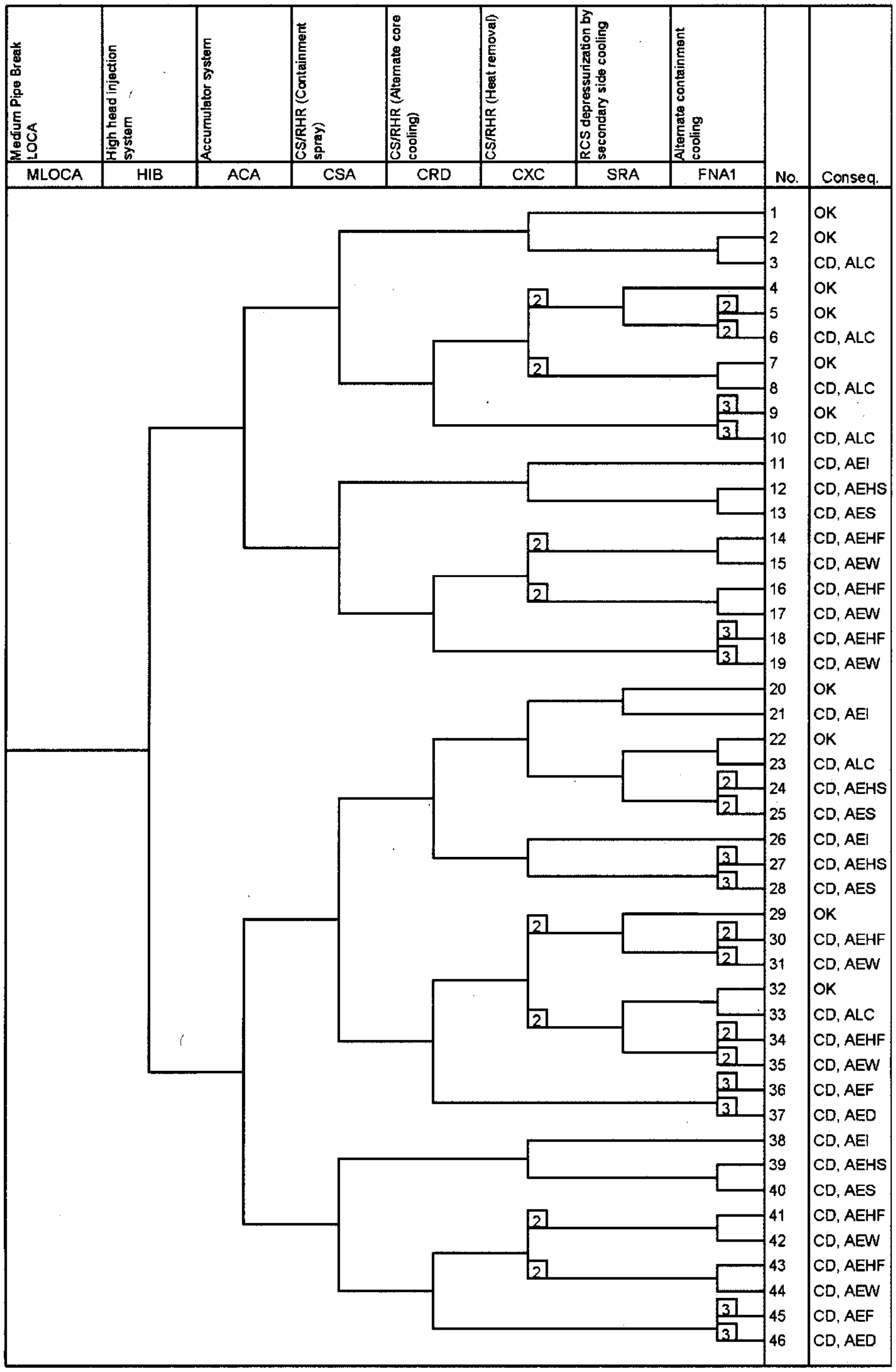

Figure 3.2.2-1 Medium Pipe Break LOCA Event Tree

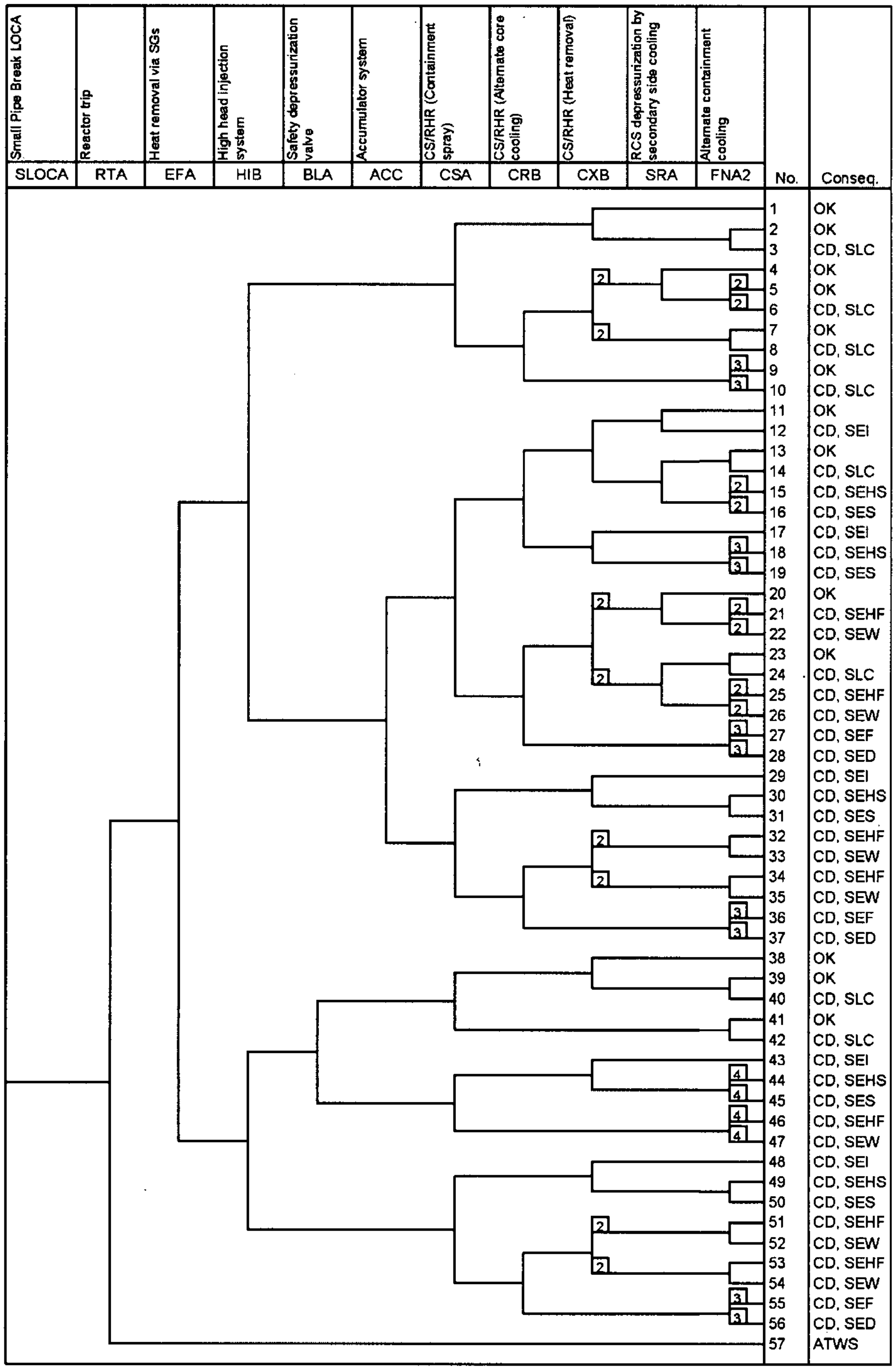

Figure 3.2.3-1 Small Pipe Break LOCA Event Tree

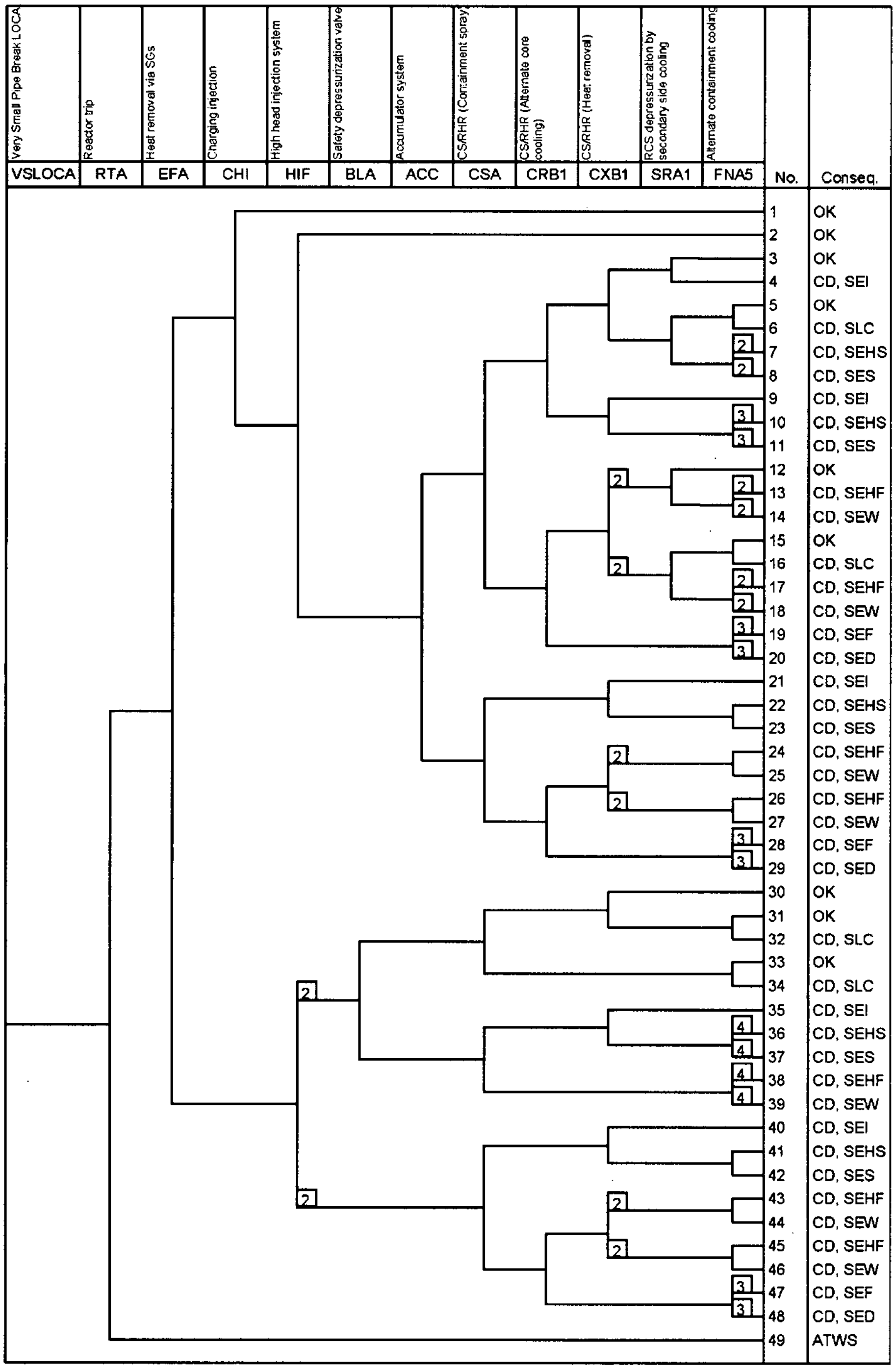

Figure 3.2.4-1 Very Small Pipe Break LOCA Event Tree

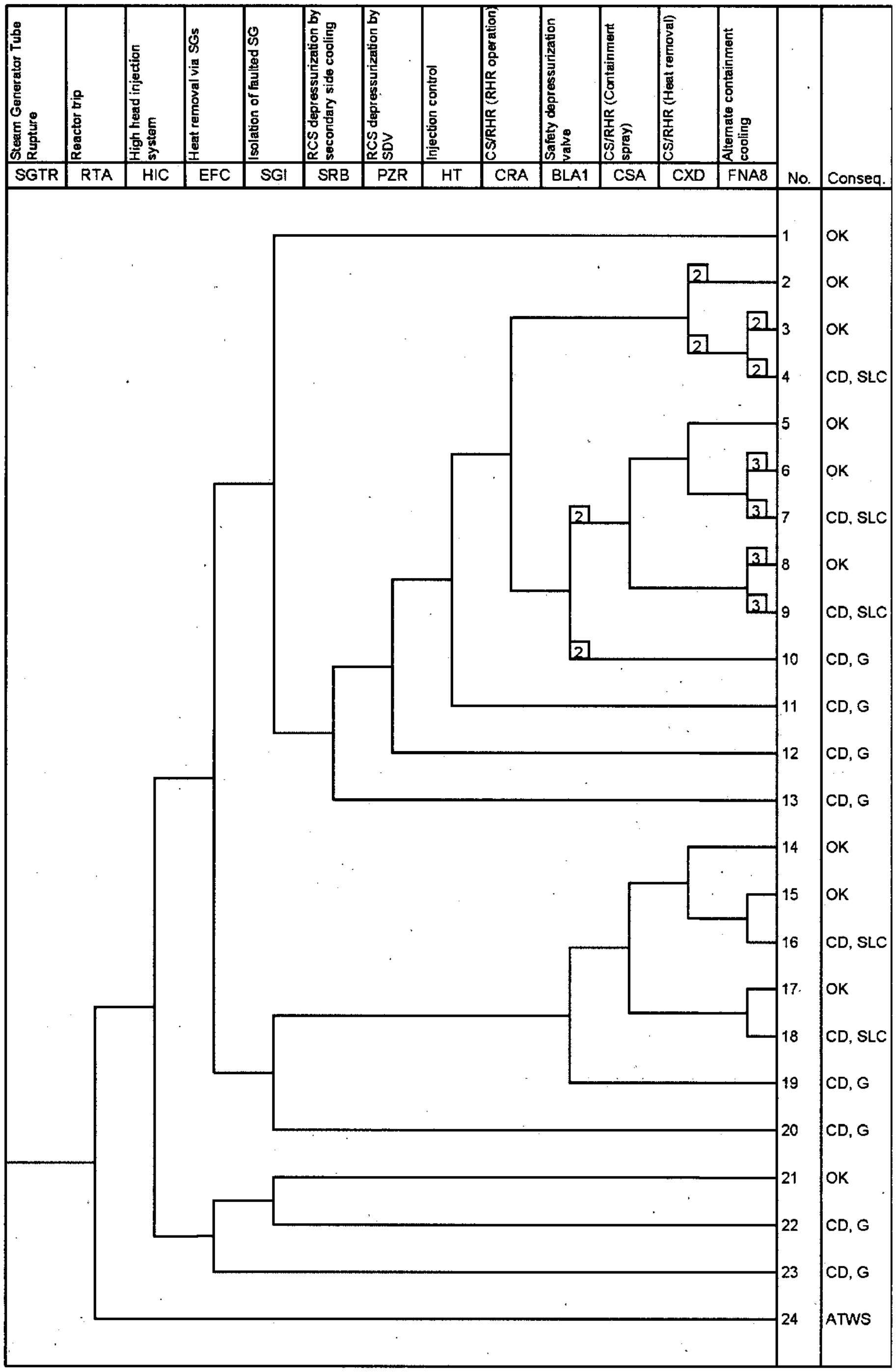

Figure 3.2.5-1 Steam Generator Tube Rupture Event Tree

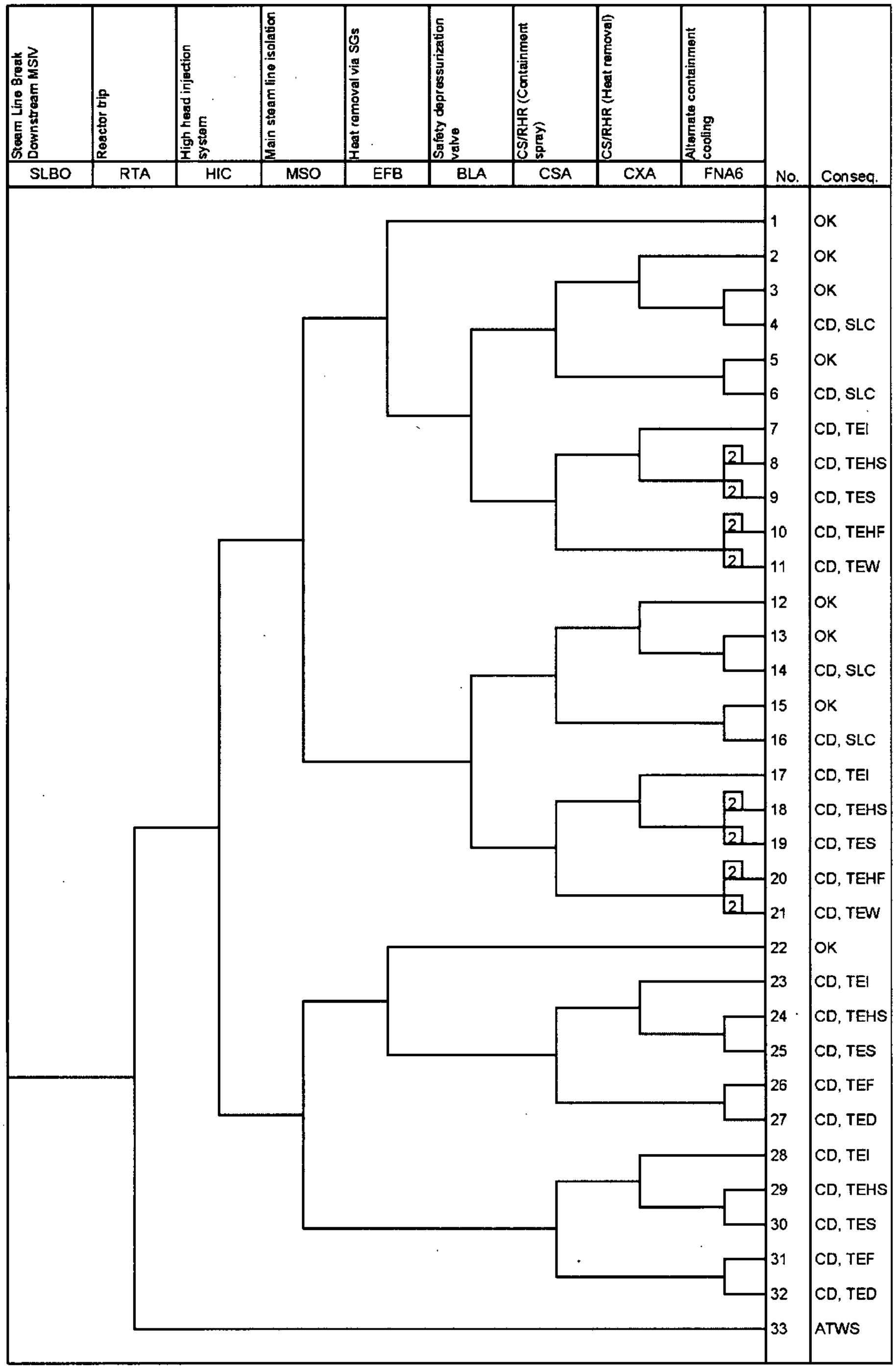

Figure 3.2.7-1 Steam Line Break Downstream MSIV Event Tree

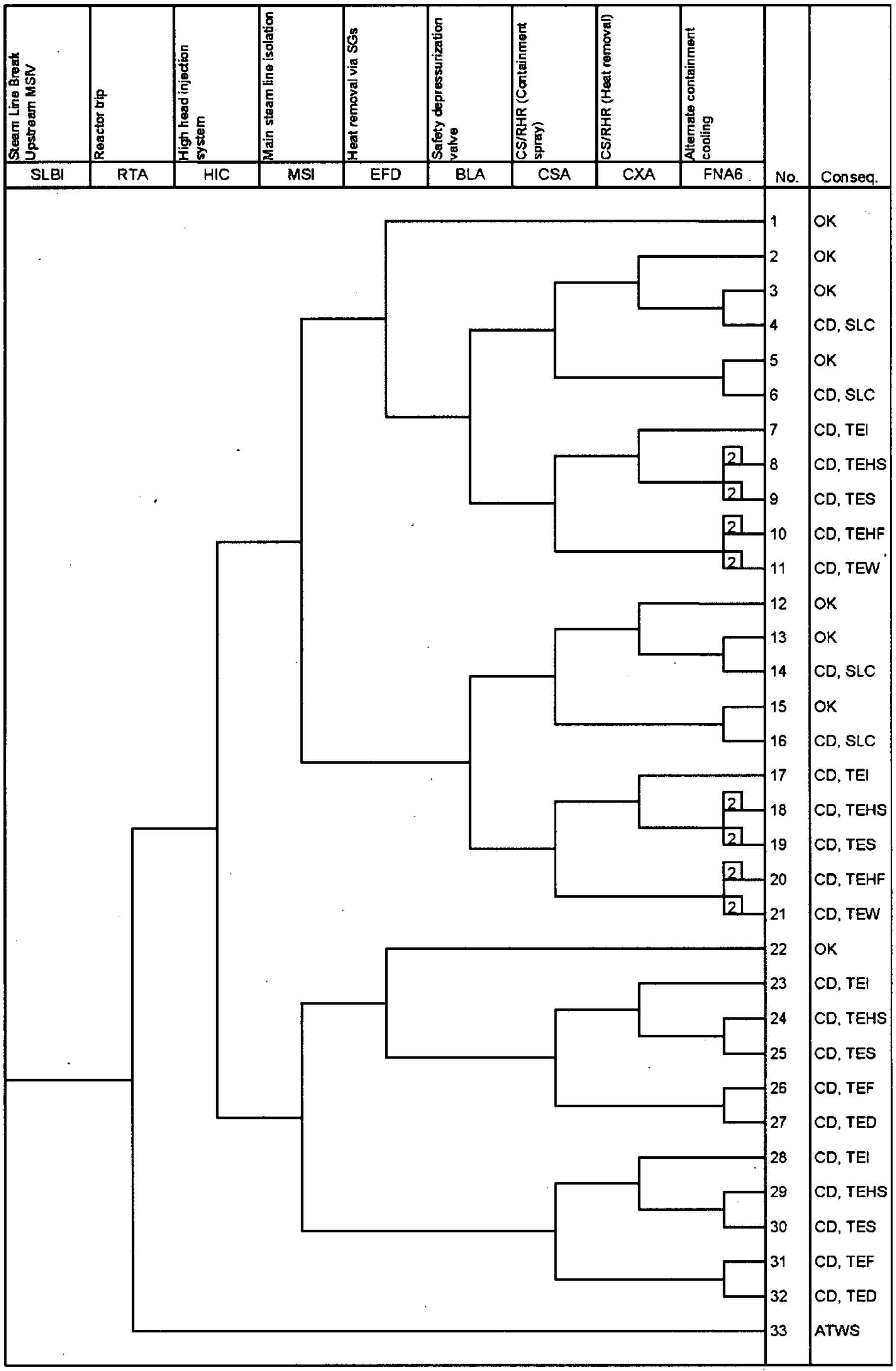

Figure 3.2.8-1 Steam Line Break Upstream MSIV Event Tree

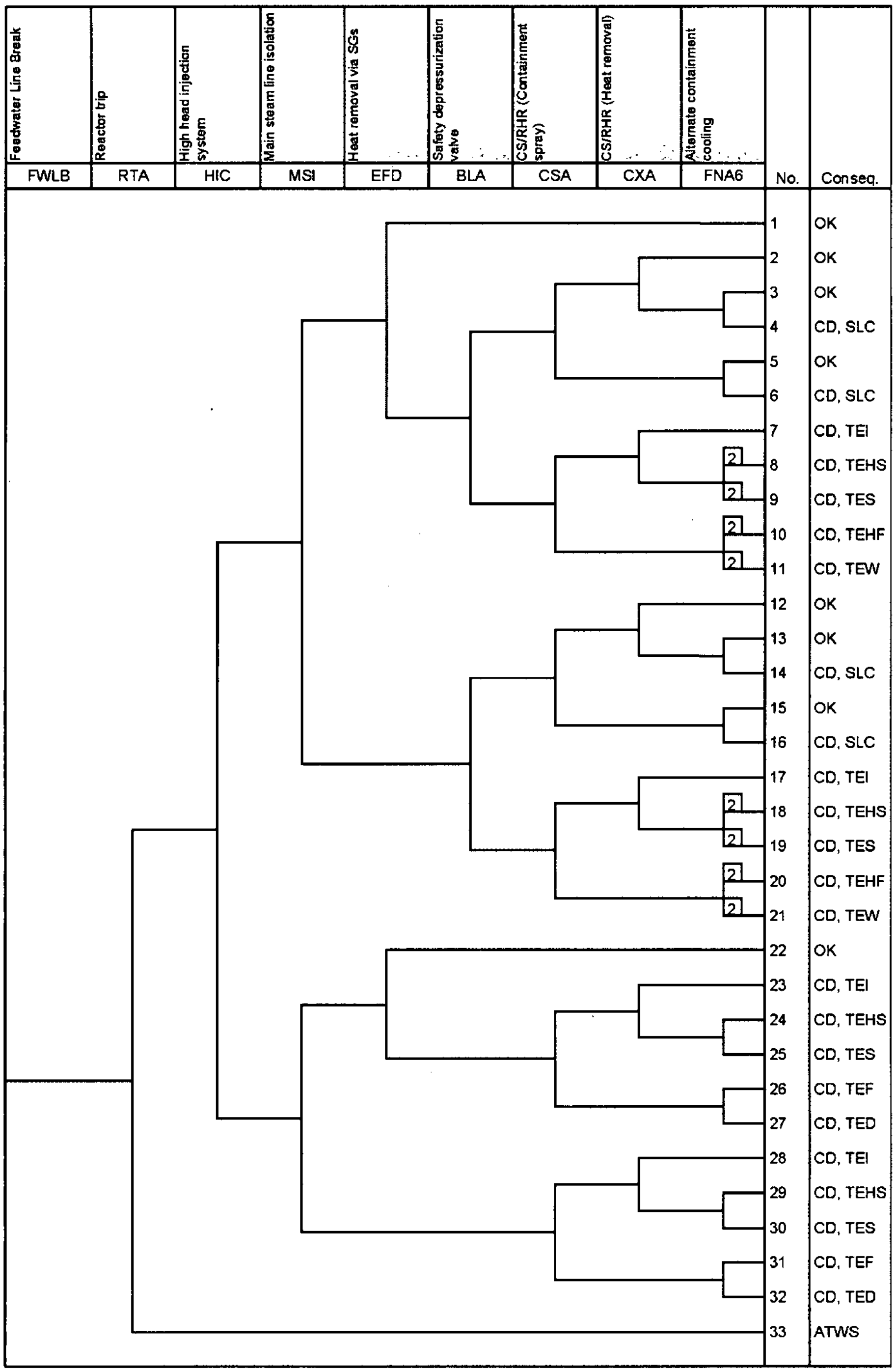

Figure 3.2.9-1 Feedwater Line Break Event Tree

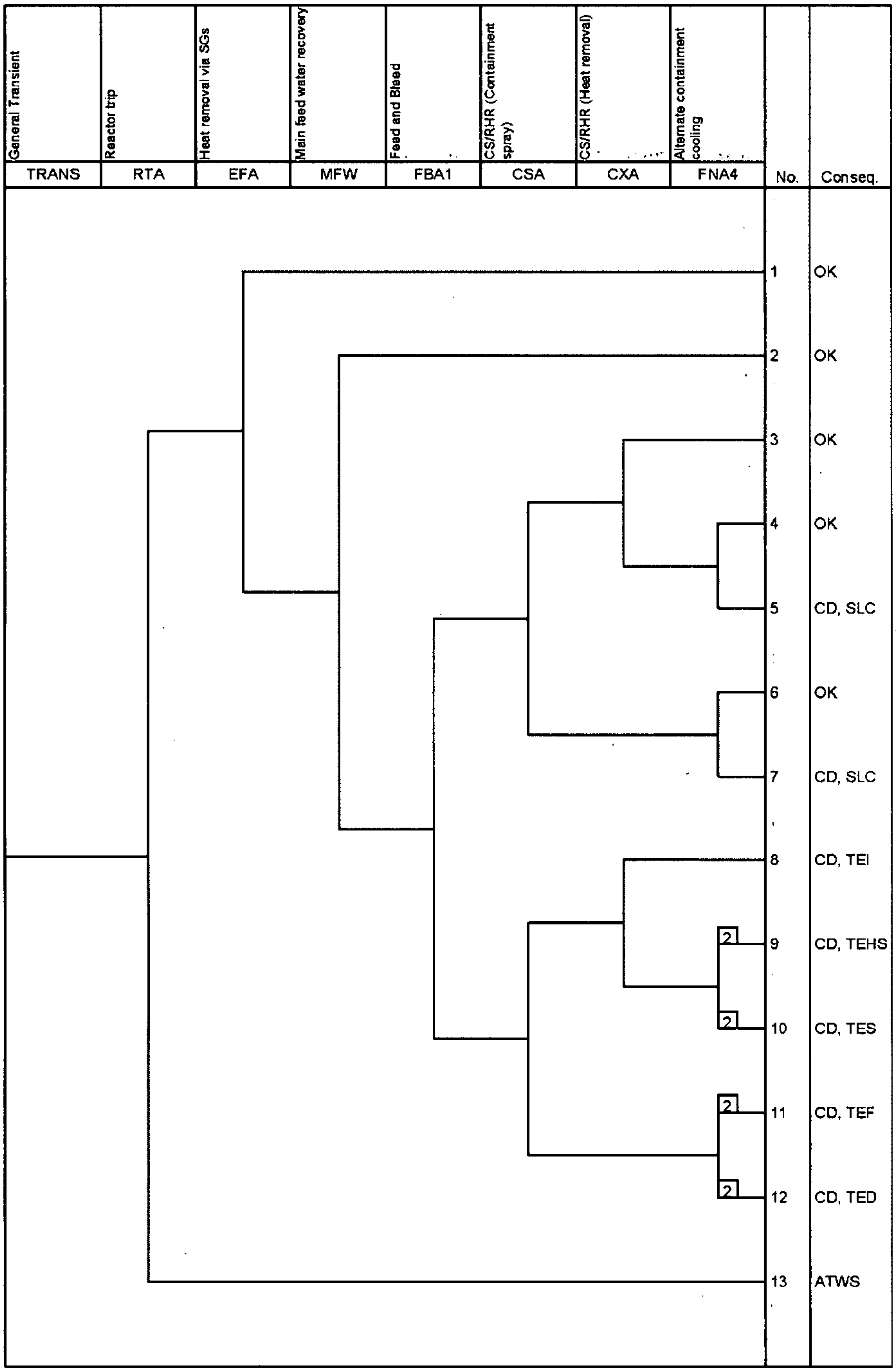

Figure 3.2.10-1 General Transient Event Tree

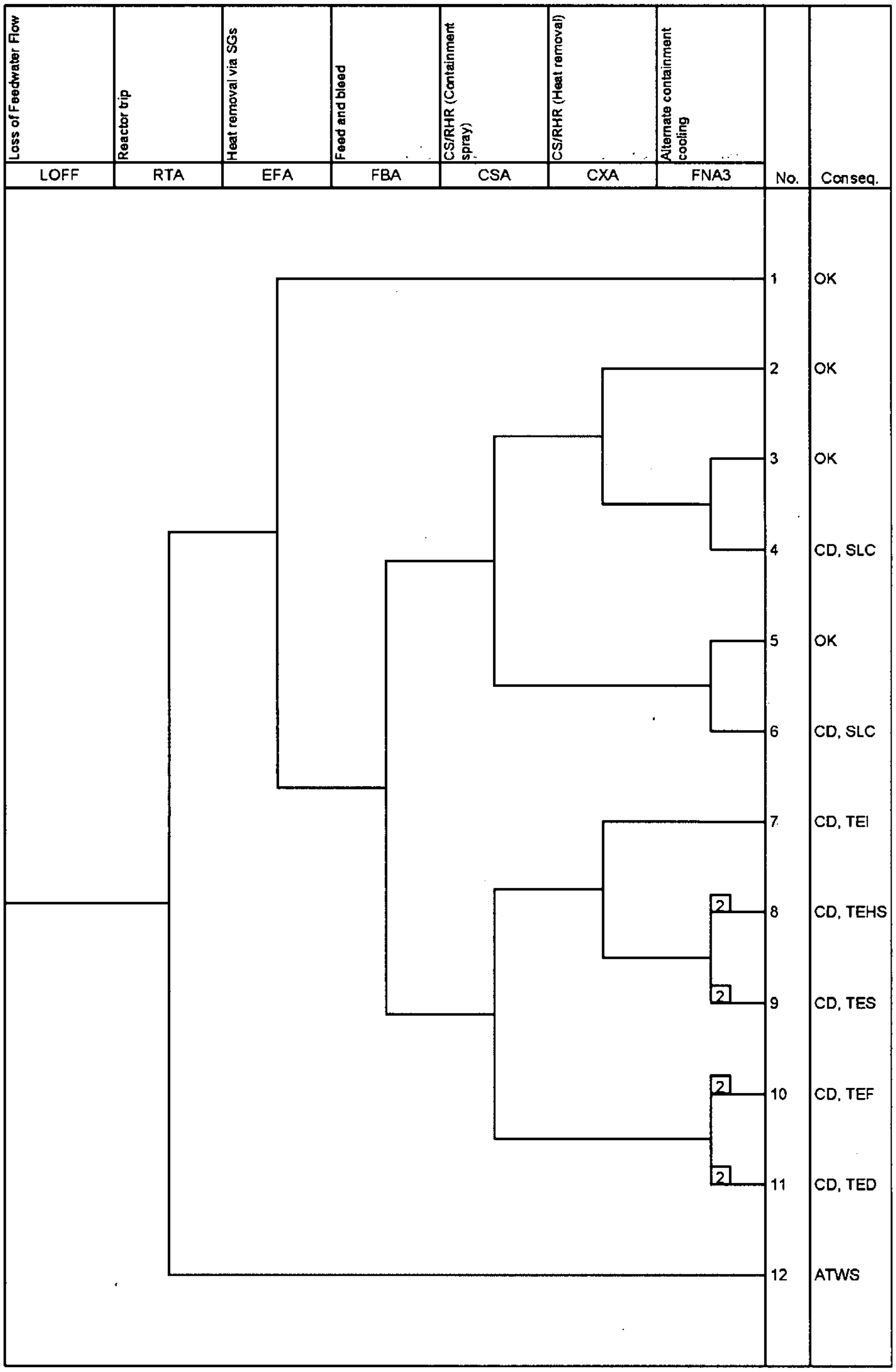

Figure 3.2.11-1 Loss of Feedwater Flow Event Tree

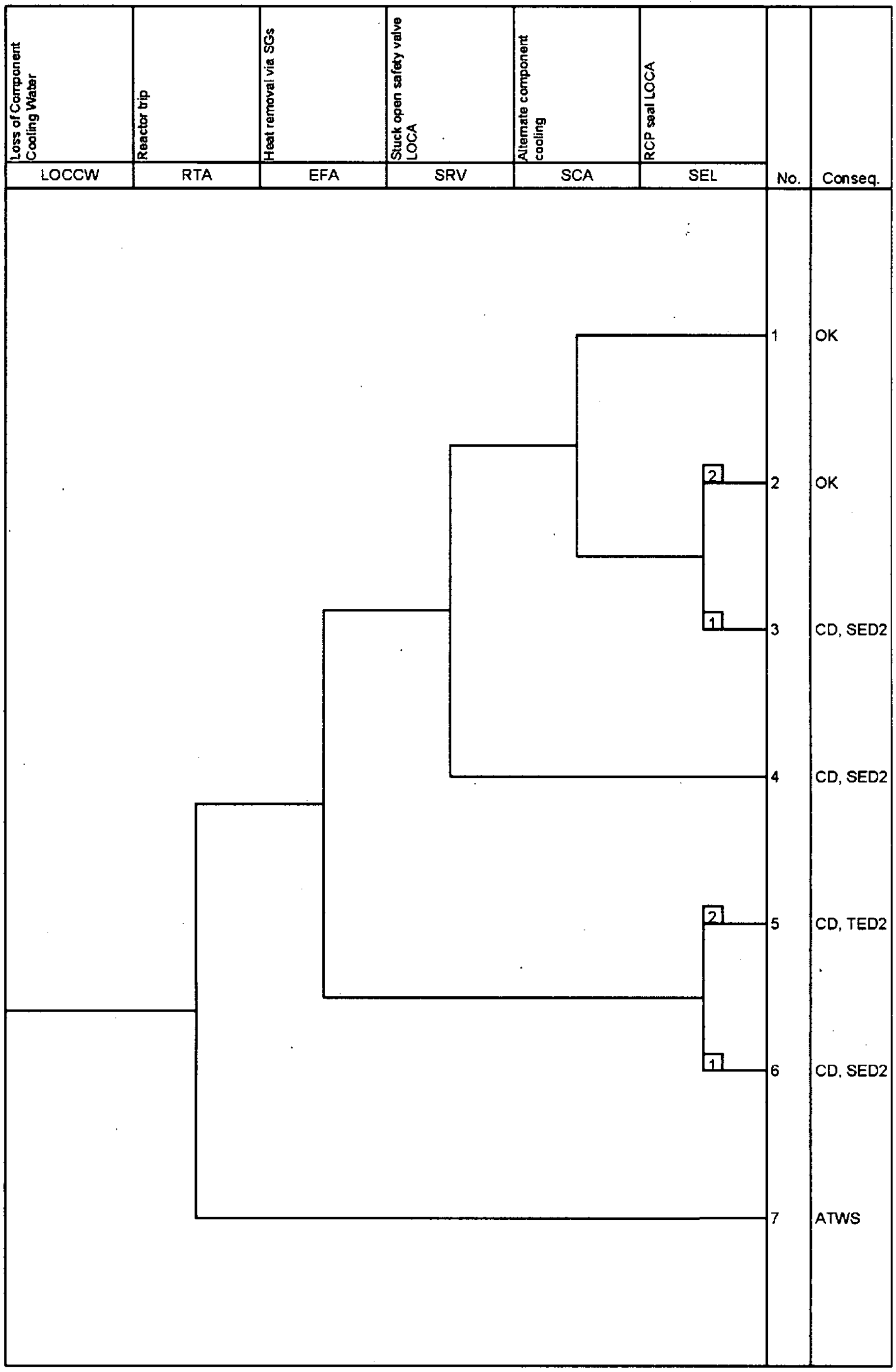

Figure 3.2.12-1 Loss of Component Cooling Water Event Tree

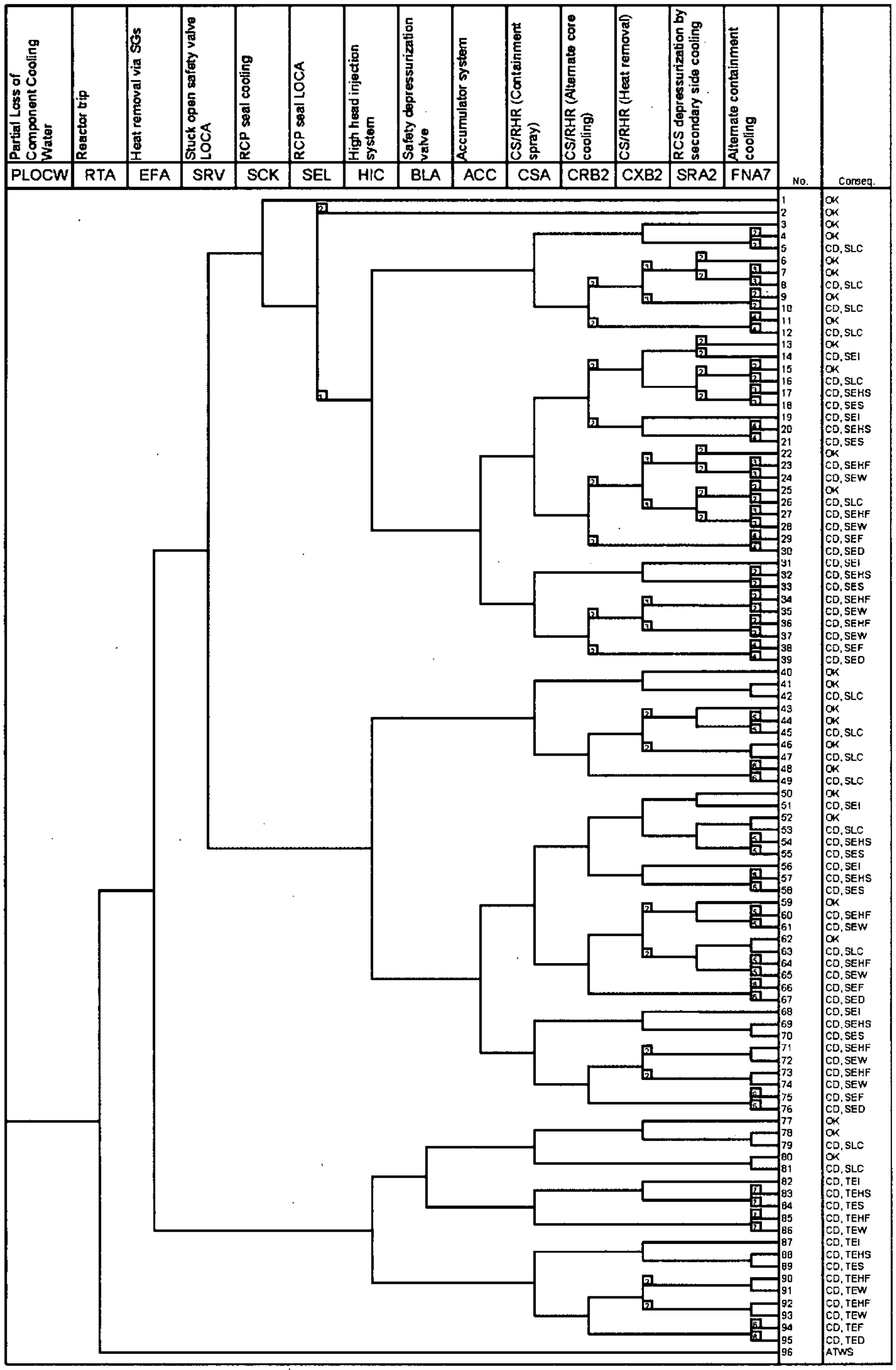

Figure 3.2.13-1 Partial Loss of Component Cooling Water Event Tree

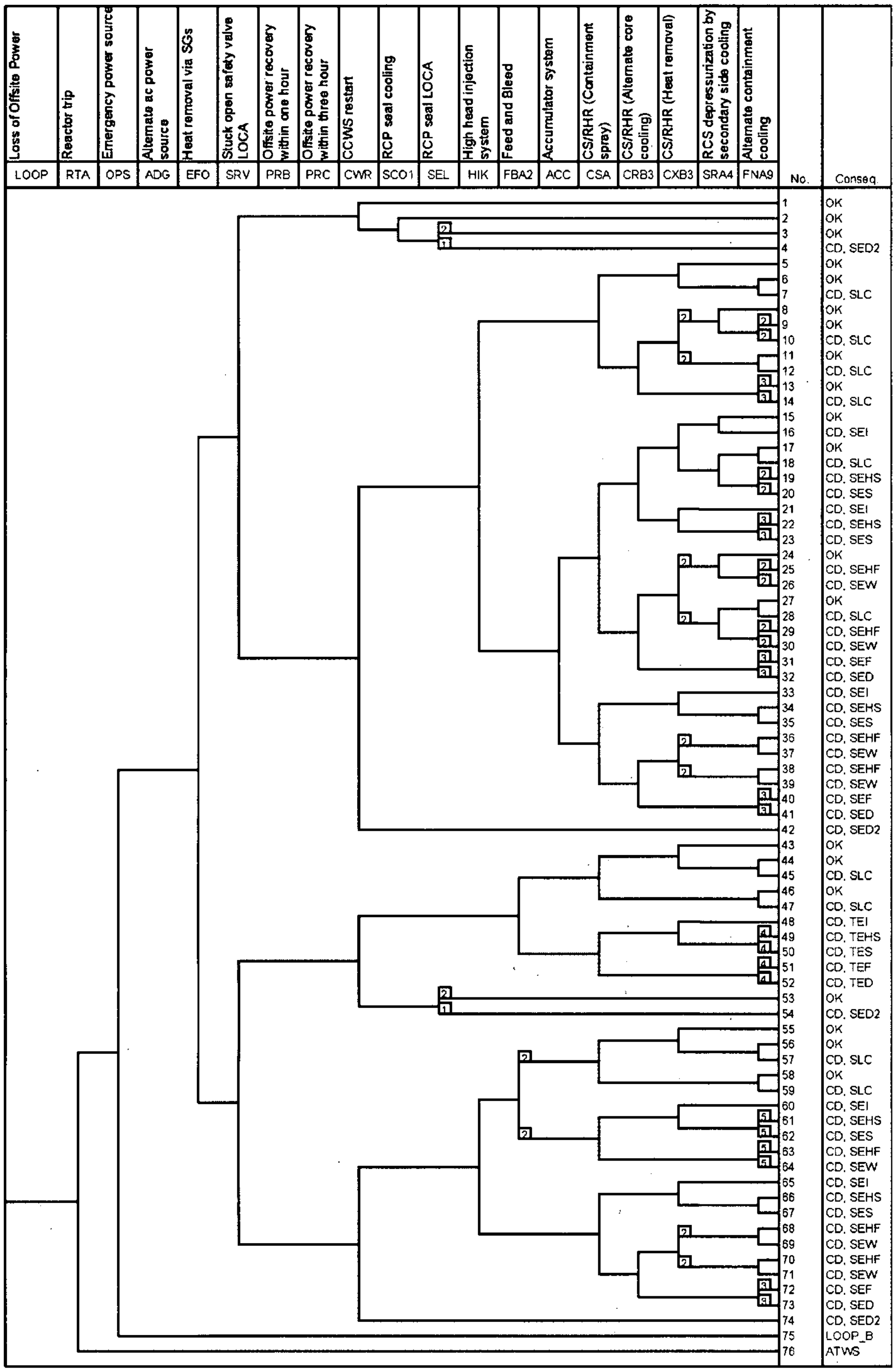

Figure 3.2.14-1 Loss of Offsite Power Event Tree (Sheet 1 of 4)

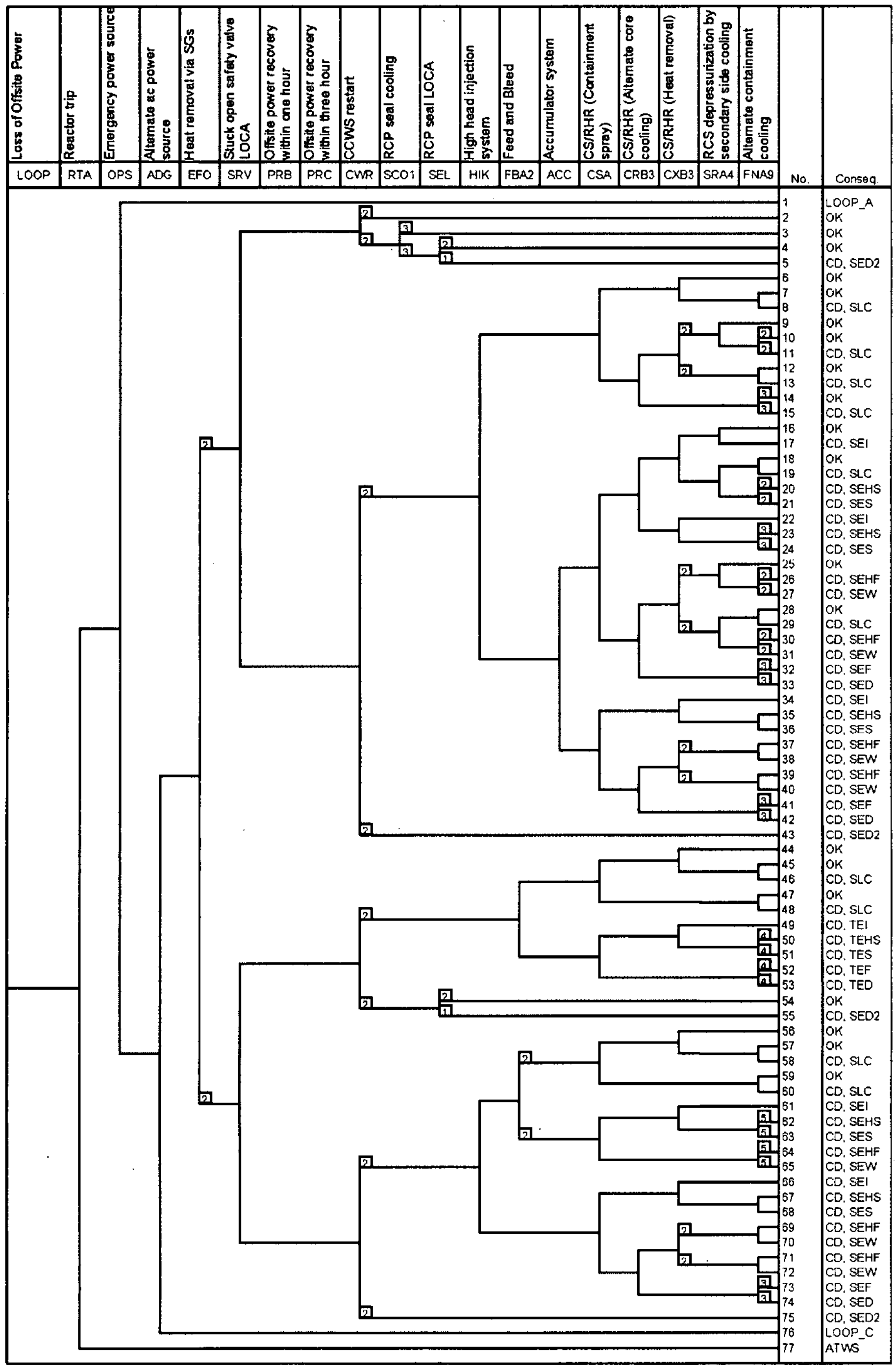

Figure 3.2.14-1 Loss of Offsite Power Event Tree (Sheet 2 of 4)

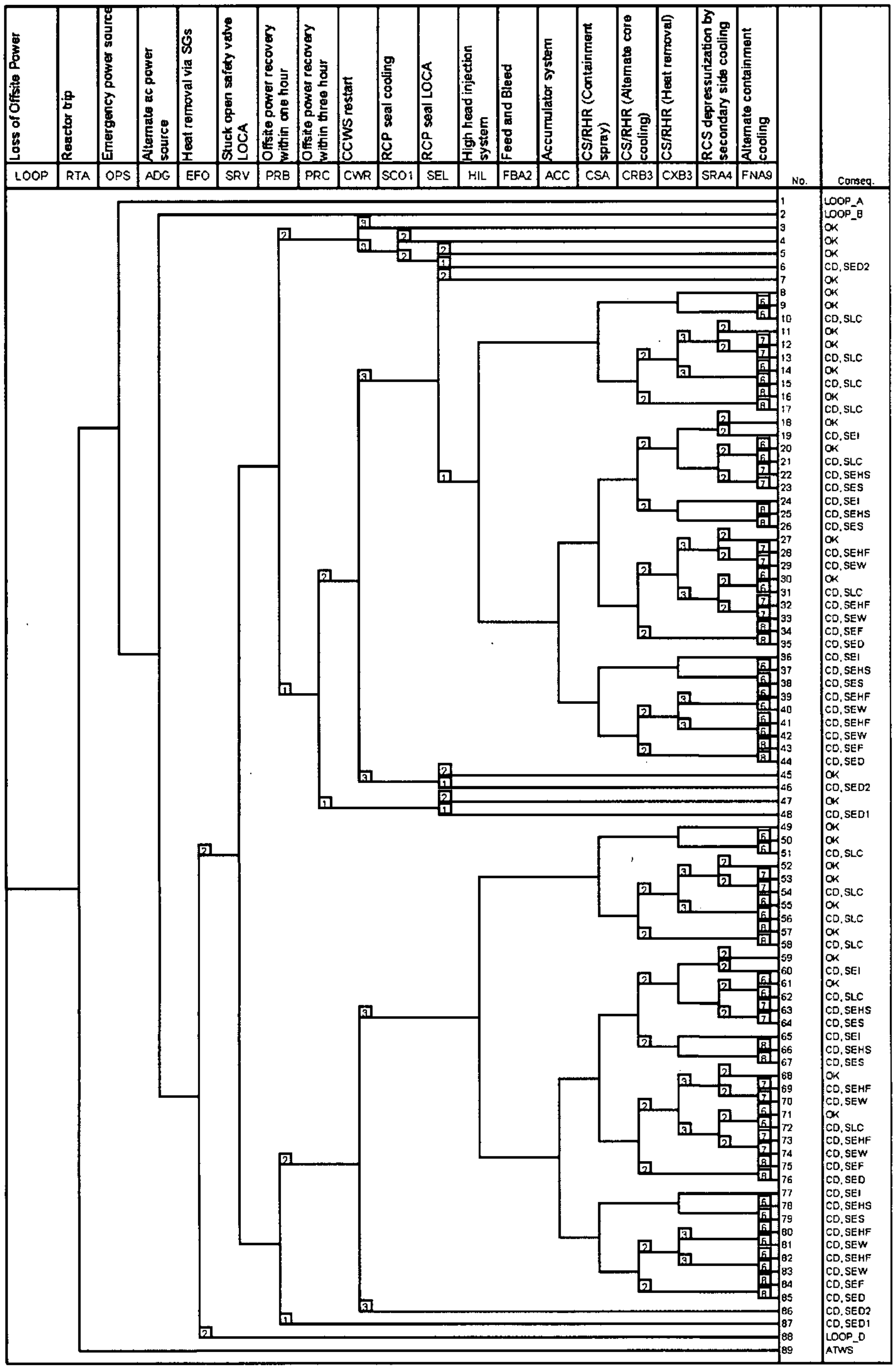

Figure 3.2.14-1 Loss of Offsite Power Event Tree (Sheet 3 of 4)

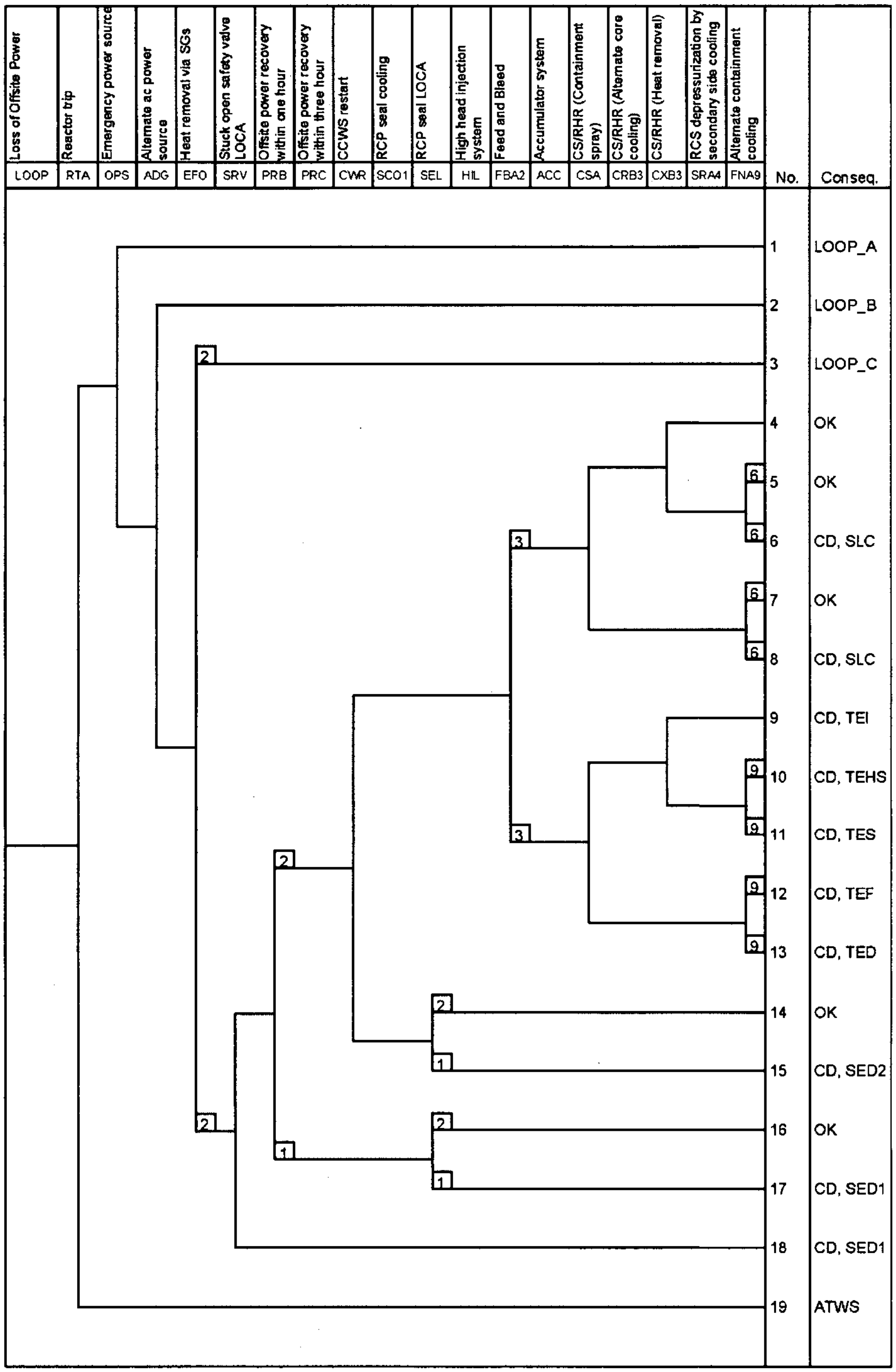

Figure 3.2.14-1 Loss of Offsite Power Event Tree (Sheet 4 of 4)

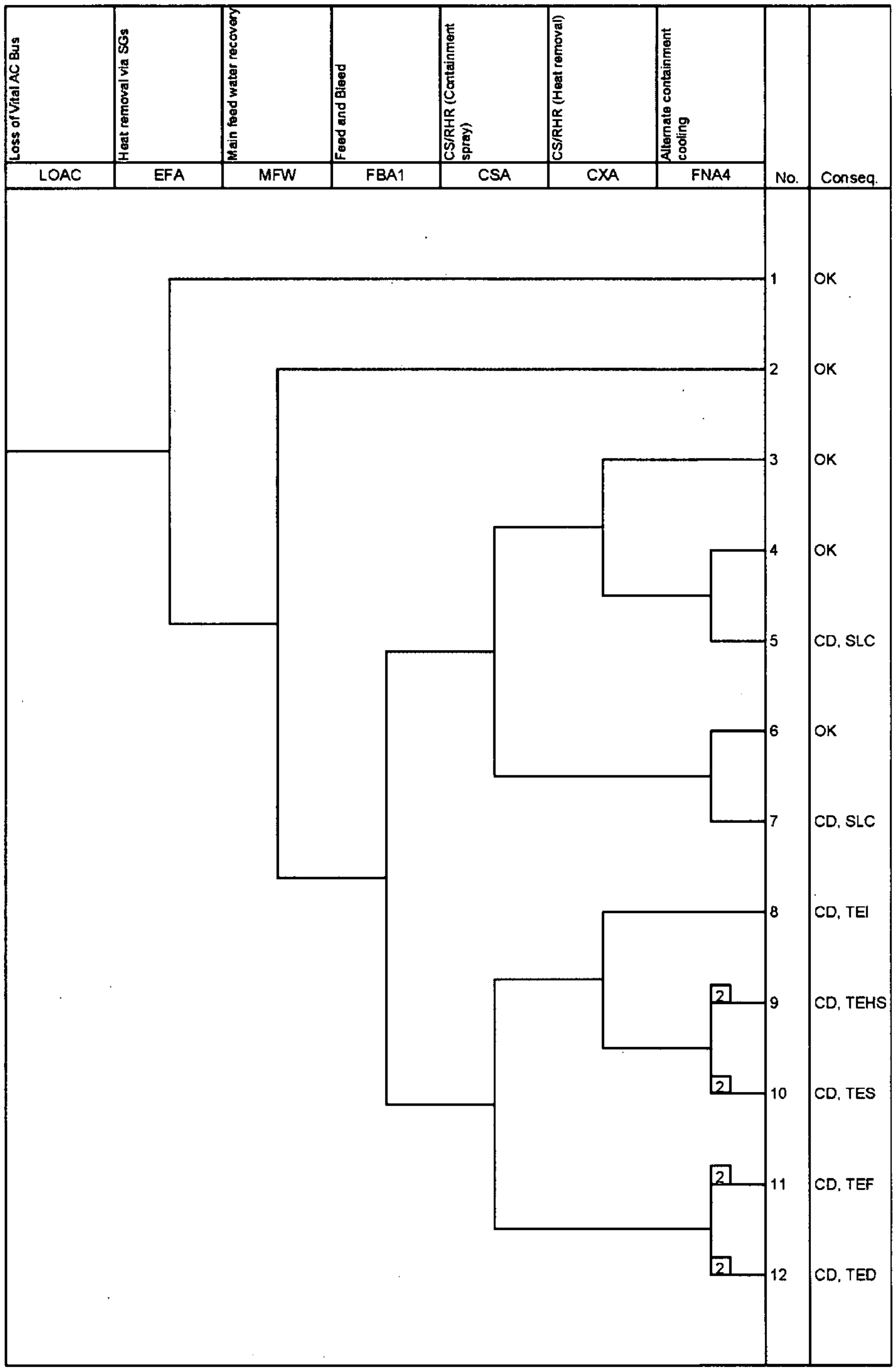

Figure 3.2.15-1 Loss of Vital AC Bus Event Tree

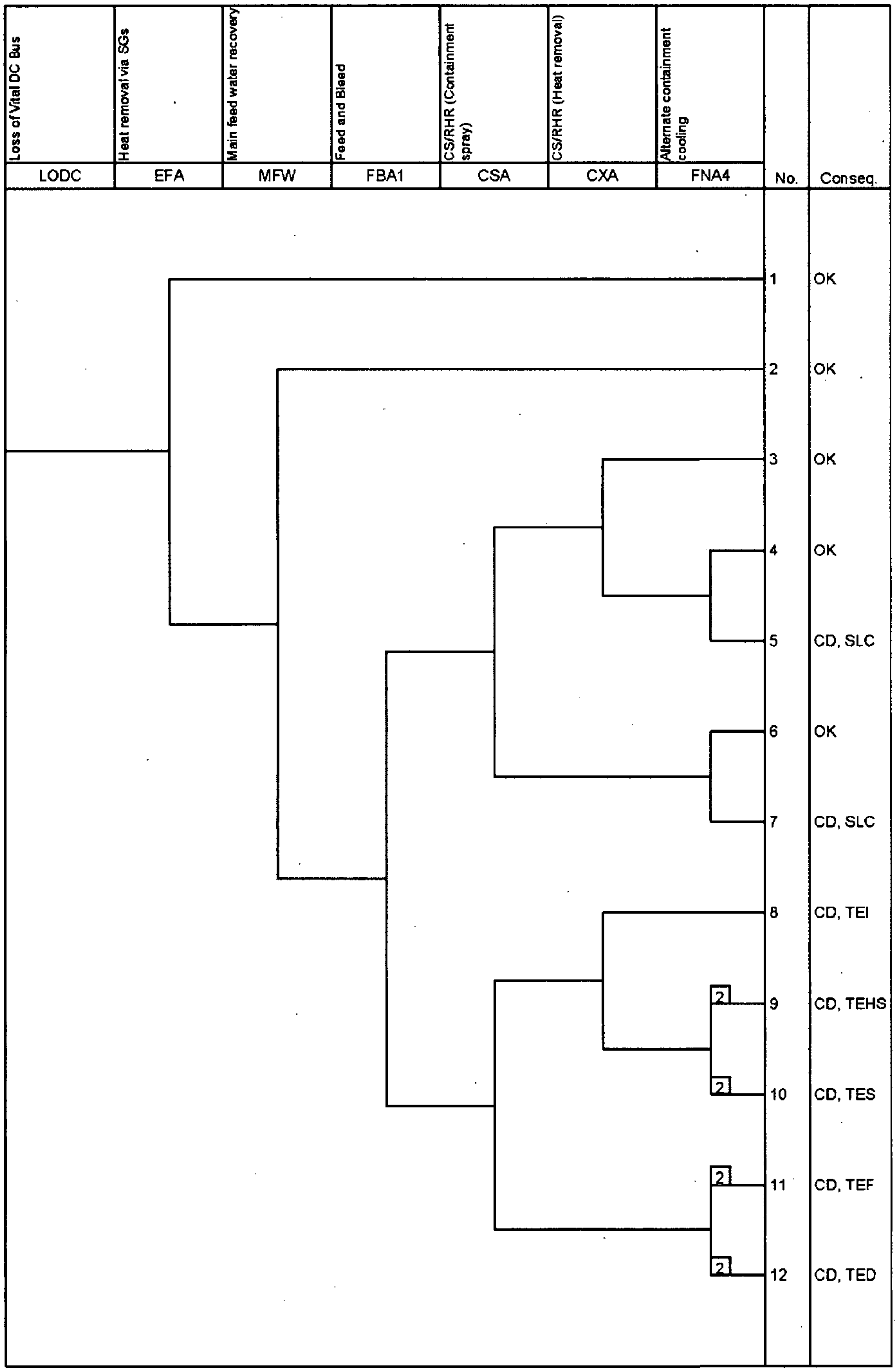

Figure 3.2.16-1 Loss of Vital DC Bus Event Tree

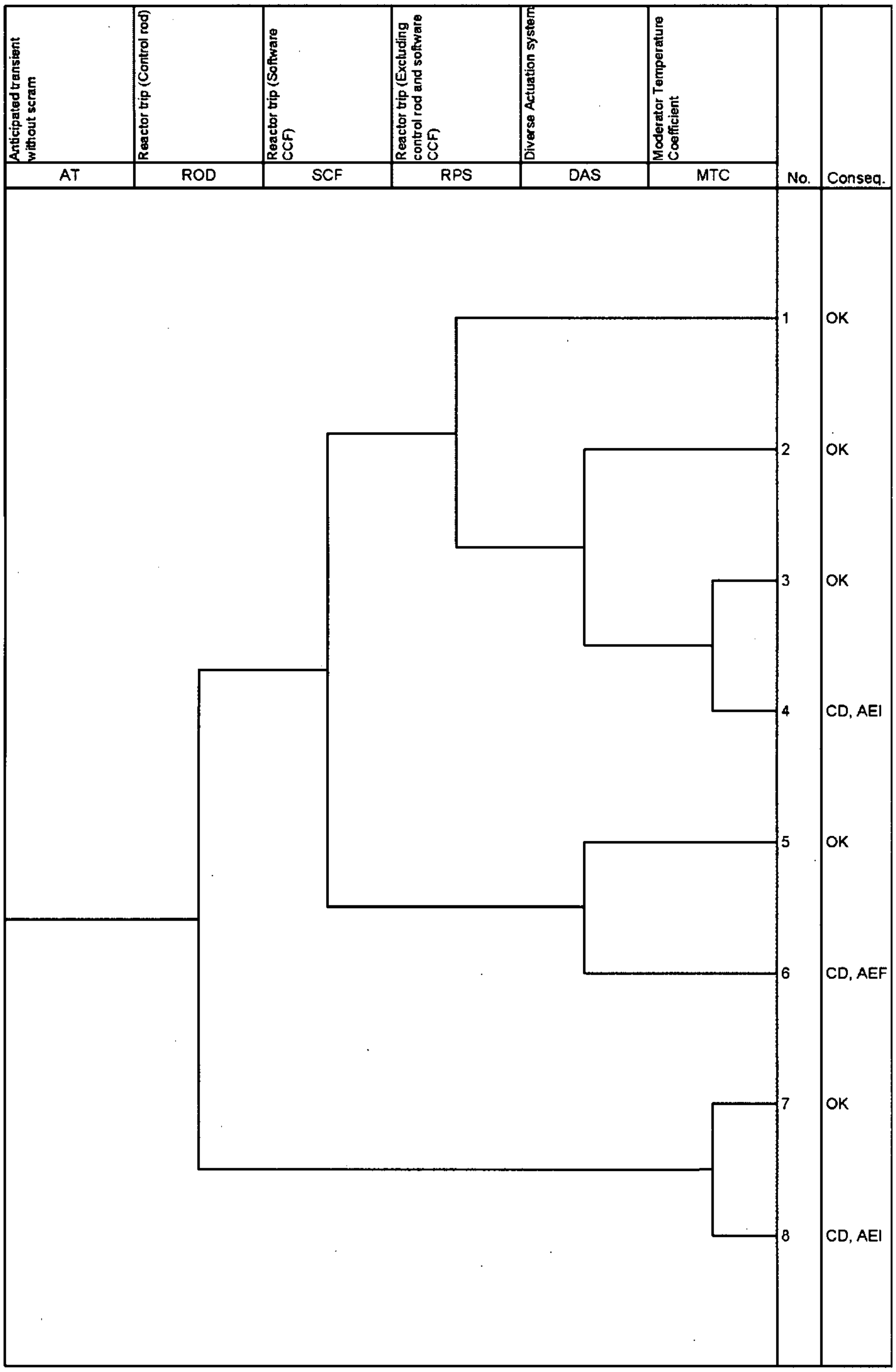

Figure 3.2.17-1 Anticipated transient without scram Event Tree