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TECHNOLOGIES INC.

June 07, 2004

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U.S. Nuclear Regulatory Commission,
Document Control Desk,
Washington, D.C. 20555

Attention: Ms. B. Sosa
Project Manager, ACR

References:

1. Letter J. Kim to V.J. Langman, "Requests for Additional Information on the Event Categorization - ACR-700 Pre-Application Review (TAC No. MB5765)", April 23, 2004.

Re: **Response to NRC's Requests for Additional Information (RAIs) on Event Categorization**

In response to NRC's request (Reference 1) and in support of the NRC's pre-application review of the ACR-700, Attachment 1 provides AECL's responses to NRC staff requests for additional information on Event Categorization.

If you have any questions on this letter and/or the enclosed material please contact the undersigned at (905) 823-9060 extension 6543.

Yours sincerely,

Vince Langman

Vince J. Langman
ACR Licensing Manager

/Attachment:

1. Response to NRC's Requests for Additional Information (RAIs) on Event Categorization

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Attachment 1
(Letter V.J. Langman to B. Sosa, "Response to NRC's Requests for Additional Information (RAIs) on Event Categorization", June 07, 2004)

Response to NRC's Requests for Additional Information (RAIs) on Event Categorization

AECL's responses to NRC's requests for additional information on Event Categorization are provided in italic fonts following each of the NRC's questions as follows. The tables referenced in AECL's responses are provided at the end of this attachment.

224. Tables 2-1 and 2-2 of Reference 1 list initiating events under Classes 1 through 5 categories. The associated acceptance criteria are summarized in Table 2-3 for the ACR-700 analysis. Please provide the following information:

    (1). Discuss the methods, plant operating experience, and test data that are used to determine the event categories. List the frequency of occurrence for each initiating event listed in Tables 2-1 and 2-2. Where applicable, please define the event categories using the frequency of occurrence for the Classes 1 through 5 events.

    (2). Compare the event category and the associated acceptance criteria for each initiating event documented in Tables 2-1 and 2-2 of Reference 1 with those specified in Chapter 15 of NUREG-0800, "Standard Review Plan (SRP)," (Ref. 2), and address the ACR-700's compliance with the SRP Chapter 15 guidance in the event categorization and the associated acceptance criteria for each initiating event. Justification for any deviations from the SRP needs to be discussed.

    (3). Compare the event category and the associated acceptance criteria for each initiating event documented in Tables 2-1 and 2-2 of Reference 1 with those specified in Tables 4.1 and 4.9 of Draft Regulatory Guide (DRG) C-006 (Reference 3), "Safety Analysis of CANDU Nuclear Power Plants," and address the ACR-700 compliance with the DRG C-006 guidance in the event categorization and the associated acceptance criteria for each initiating event. Justification for any deviations from DRG C-600 needs to be discussed.

References:
1. ACR-700 10810-03510-AB-001 (Rev. 0), "Initial Conditions and Standard; Assumptions Safety Analysis Basis", dated August 14, 2003.
2. NUREG-0800 (Revision 2), "Standard Review Plan", dated July 1981.
3. Draft Regulatory Guide C-006 (R1(E)), "Safety Analysis of CANDU Nuclear Power Plants", dated September 1999.

AECL Response:
224. (1)
Event Categorization:
The ACR-700 approach for classification of events is provided in the ACR safety basis document (Reference [A1]). In this approach, the higher probability events are required to meet more restrictive dose limits. The approach, taking into account the pressure tube design, uses three categories of events: design basis accidents, limited core damage accidents, and severe core damage accidents. Event frequency ranges of the three event categories are presented in Table A1.

The event categories are determined based on engineering judgment taking into account operating experience and relevant analyses. By applying a systematic review to the ACR-700 plant, initiating events are identified and categorized into appropriate classes.

With respect to US licensing of the ACR-700, AECL intends to meet the US requirements for design basis accidents as discussed in the position paper (Reference [A2]) that has been recently submitted to the NRC. For design basis accidents for US Light Water Reactors (LWRs), the NRC has subdivided the design basis accidents into three event classifications as published in Regulatory Guide 1.70 (Reference [A3]):

- *Incidents of moderate frequency - these are incidents, any one of which may occur during a calendar year for a particular plant.*
- *Infrequent incidents - these are incidents, any one of which may occur during the lifetime of a particular plant.*
- *Limiting faults - these are occurrences that are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.*

For the ACR-700, the initiating event frequency would still be used to determine the event classification, corresponding to the three categories from Regulatory Guide 1.70. This results in a system of classification of design basis accidents for the ACR-700 that is generally consistent with US light water reactor practices, i.e., the initiating event frequency determines the classification. Dose consequences for design basis accidents for the ACR-700 will be evaluated against an increasing set of allowables, but in all cases the consequences are within 25 Rem total effective dose equivalent (and are substantially less than 25 Rem for the higher frequency events).

The limited core damage accidents (i.e., Classes 4 and 5 events) of the ACR safety basis are an intermediate category which does not have an equivalent in the US practice. The Limited Core Damage Accidents (LCDAs) are accidents strictly and solely tied to the pressure-tube reactor design of the ACR. They consist of improbable occurrences (therefore beyond the design basis) involving either significant damage of the fuel in a single channel due to the failure of a process condition in the channel itself (severe flow blockage or stagnation feeder break) or an

overheating of the whole core due to loss of the normal and engineered emergency heat sinks. In light of this, AECL has prepared a position paper (Reference [A2]) to the NRC, proposing a treatment of limited core damage accidents with respect to design certification of the ACR-700 in the US.

The severe core damage accidents in the ACR safety basis are the direct equivalent of the severe accidents in the US practice: all such accidents are beyond the design basis and involve the ultimate disassembly of the core and the consequential loss of core coolability.

Event Frequency:
A preliminary systematic review of plant design for identification of initiating events (Reference [A4]) for the ACR-700 design has been performed. Table A2 provides the frequencies of the design basis accidents listed in Table 2-1 of Reference [A5]. The frequencies of occurrence that are not yet available are listed as "to be determined" ("TBD"). These preliminary frequency values are based on the probabilistic safety assessment (PSA) work performed on the existing CANDU plants.
The frequencies of occurrence for the LCDAs are not available at the current design stage. The LCDAs also include an initiating event combined with the unavailability of the emergency core cooling (ECC) system. The ECC system is designed to meet the short-term unavailability target of 1.0E-3/yr. An example of the frequency of a combined event, such as a large LOCA (i.e., 4.18E-04/yr) combined with a loss of ECC system (i.e., 1.0E-3/yr), will be approximately in the range of 1.0E-7/yr.

AECL Response:
224. (2)
Table A3 provides a mapping of the events listed in Table 2-1 of Reference [A5] with the relevant chapters of the SRP for Chapter 15, along with a comparison of the corresponding acceptance criteria. The event categorization for certain events has been changed somewhat from that in Reference [A5]. The change reflects that the event class for combinations of initiating event with Loss of Offsite Power or Loss of Class IV Power is the same as the event class for the initiating event. The reply to RAI number 226, explains the change in more detail. In general, dose limits provided in Reference [A5] are more restrictive than dose limits given in the SRP. Since we will meet the more restrictive dose limits, the deviation from SRP requirements is in the conservative direction. ACR-700 design certification submissions will show compliance of the Total Effective Dose Equivalent (TEDE) with the SRP dose limits.
ECC acceptance criteria from the SRP will be demonstrated for the limiting large LOCA case. For the single channel events, compliance with the ECC acceptance criteria will be demonstrated for the intact channels.
For each of the events, the acceptance criteria are either the same as the SRP or are more conservative.
The events in Table A4 (Table 2-2 of Reference [A5]), which are beyond design basis accidents, have no equivalent in the SRP Chapter 15, therefore they have not been included in the table. Reference [A2] provides the rationale for the classification and proposes an approach to the treatment of this class of accidents.

AECL Response:
224. (3)
As discussed in the response to RAI 224(1), the intent is to follow the US approach for design basis accidents.
The Canadian Nuclear Safety Commission (CNSC) is in the process of preparing a licensing basis for advanced reactors in Canada, which may affect event classifications and acceptance criteria. Therefore a detailed comparison of Reference [A5] with C-006 would likely not be significant at this stage.
The Limited Core Damage Accidents provided in Table A4 (Table 2-2 of Reference [A5]) are a category of beyond design basis accidents and they consist primarily of the Class 4 and 5 accidents. The Stagnation Feeder Break and the Severe Channel Flow Blockage are now classified as LCDAs in Reference [A5]. These accidents are classified as LCDAs because of their low probability and are unique to the pressure-tube reactor. The rationale for the reclassification is given below.
The Stagnation Feeder Break results in flow stagnation in a single channel and consequential channel rupture. While the likelihood of achieving the precise break flow area required to produce a stagnation feeder break is very low, the physical possibility is acknowledged as a LCDA. Feeder breaks outside the stagnation break range will not result in fuel channel rupture. In all feeder break cases, any predicted fuel damage is restricted to a single channel.
Similarly, a Severe Flow Blockage represents a small range of blockage areas that would result in channel rupture. Only blockages that reduce the flow area by approximately 90% or more will result in channel failure. The probability of such a blockage occurring is correspondingly low. Any predicted fuel damage is restricted to the single affected channel for the whole range of blockage areas.

References:
[A1] AECL Report 108-03600-AB-003, "Safety Basis for ACR".
[A2] V. J. Langman, "Regulatory Treatment of Limited Core Damage Accidents for the ACR-700", AECL Letter to US NRC, May 10, 2004.
[A3] Regulatory Guide 1.70, "Standard Format and content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)", Revision 3, November 1978.
[A4] AECL Report 108-03660-ASD-001, "Systematic Review of Plant Design for Identification of Initiating Events".
[A5] AECL Report 10810-03510-AB-001, "Initial Conditions and Standard Assumptions Safety Analysis Basis".

225. The requirements of loss-of-coolant-accidents (LOCAs) analyses for light water reactors are documented in 10 CFR 50.46 (Ref. 4), which requires that the LOCA analyses consider different postulated break sizes and locations. The results of the analyses must satisfy the following acceptance criteria: (1) PCT less than 2200 ^{0}F; (2) maximum local oxidation of the cladding less than 17% of the total cladding thickness; (3) hydrogen generated from the chemical reaction of the cladding with water or steam less than 1 percent of total metal in the cladding cylinders surrounding the fuel; (4) no loss of core cooling capability; and (5) long term cooling capability.

Please address ACR's compliance with the 10 CFR 50.46 requirements regarding the scope of LOCA analyses and the acceptance criteria. Discuss any resolutions in the situation where the 10 CFR 50.46 requirements are not met.

Reference:

4. CFR50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-water Nuclear Power Reactors".

AECL Response:
The LOCA analysis to be reported during ACR-700 design certification will demonstrate compliance with 10 CFR 50.46 acceptance criteria. The analysis will cover the spectrum of break sizes and consider appropriate break locations, including the Reactor Inlet Header (RIH), Reactor Outlet Header (ROH), pump suction pipe, Emergency Coolant Injection (ECI) line and reactor outlet header large interconnect pipe.
For single channel events, the analysis will demonstrate compliance with the acceptance criteria for fuel in the non-affected channels.

226. Table 2-1 of Reference 1 indicates that some design-basis-events (DBEs) (such as single steam generator (SG) tube rupture) consider the licensee vendor inspection program (LCVIP) and some (such as Class 1 events) do not. When the DBEs consider LCVIP, the event class recategorizes to the next higher class for some combined events. For example, the partial single channel flow blockage is a class 2 event while the same event with a combination of LCVIP becomes a class 3 event. For some other DBEs (such as main steam line break), the event class remains unchanged for cases with and without LCVIP.

(1). Please discuss the criteria that are used to determine the inclusion of LCVIP in the DBE analysis and to recategorize the combined events, and identify the causes that may result in LCVIP.

(2). With regard to the requirements of loss of offsite power (LOOP) analysis, please address the ACR-700's compliance with the requirement of General Design Criteria (GDC)-17 (Ref. 5) that states, in part, that: "... An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences (AOOs) and (2) the core is cooled and containment and other vital functions are maintained in the event of postulated accidents."

Appendix A to 10 CFR 50 (Ref. 6) defines AOOs as those conditions of normal operation which are expected to occur one or more times during the life of a nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

In accordance with the GDC 17 requirements, a loss of offsite power (LOOP) must not be considered as single-failure event and must be assumed in the analysis for each of the design-basis AOOs (that encompass Classes 1 and 2 events) and postulated accidents (that encompass Classes 3 events and some of the Classes 4 and 5 events) without changing the event category.

References:

1. ACR-700 10810-03510-AB-001 (Rev. 0), "Initial Conditions and Standard; Assumptions Safety Analysis Basis", dated August 14, 2003.
5. General Design Criterion (GDC) 17, "Electric Power System".
6. Appendix A, "General Design Criteria for Nuclear Power Plants", to Code of Federal Regulation (CFR) 10, Part 50, "Domestic Licensing of Production and Utilization Facilities".

AECL Response:
The approach to change event class when combined with loss of offsite power (LOOP) is pertinent only to the Canadian licensing. This is not applicable to the US licensing. The ACR approach on US licensing is presented in a position paper (Reference [A2]) that has been recently submitted to the NRC. Nevertheless, the response on changing event category is provided for background information only.

It should be noted the consequential loss of "Class IV power" implies "offsite power". In order to be consistent with the US terminology, a consequential loss of offsite power terminology is adopted. The design basis accidents that result in a reactor trip or turbine trip such as a loss of coolant accident or main steam line break could result in a consequential loss of offsite power. One of the causes that may result in the consequential loss of offsite power is due to disruption of the grid following a sudden loss of a power generation as a result of a turbine trip. Another cause could be the failure in transferring some of the electrical loads from the unit service transformer to the station service transformer when the turbine begins to run back after a reactor trip.

The frequency of occurrence of the consequential loss of offsite power is factored into the event classification, such that when an event in Class 1 or Class 2 is combined with the consequential loss of offsite power, the resulting frequency of occurrence is in the frequency range of a higher class. Since Class 3 is the highest class in the design basis event category, a Class 3 event combined the consequential loss of offsite power is conservatively considered in the same class.

With respect to US licensing for design basis accidents, the ACR-700 design intends to comply with the safety analysis assumptions stipulated in GDC-17. As discussed in the position paper (Reference [A2]), a loss of offsite power is not considered as a single failure and must be assumed in the analysis for design basis accidents without changing the event category. Acceptance criteria are provided in the standard review plan sections for design basis accidents.

The LOOP assumption does not apply to the limited core damage accidents because these accidents are beyond design basis and are analyzed differently. The proposed treatment of the LCDAs is presented in the position paper (Reference [A2]).

Reference:
[A2] V. J. Langman, "Regulatory Treatment of Limited Core Damage Accidents for the ACR-700", AECL Letter to US NRC, May 10, 2004.

227. Table 2-2 of Reference 2 indicates that limits core damage accident consists of initiating events with a combination of loss of emergency core coolant (LOECC). Please identify the causes that may result in LOECC and confirm whether a single failure event could result in LOECC or not.

As defined in Appendix A to 10 CFR 50 (Ref. 6), "a single failure means an occurrence which results in the loss of capability to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components re function properly) nor (2) a single of a passive component (assuming active components function properly), results in loss of the capability of the system to perform safety functions."

References:
2. NUREG-0800 (Revision 2), "Standard Review Plan", dated July 1981.
6. Appendix A, "General Design Criteria for Nuclear Power Plants", to Code of Federal Regulation (CFR) 10, Part 50, "Domestic Licensing of Production and Utilization Facilities".

AECL Response:
The Emergency Core Cooling (ECC) system is designed to meet Appendix A of 10 CFR 50. The ECC system design will be assessed to demonstrate compliance with the single failure criterion. In the ACR-700 design, there is no single active component failure that can result in failure of the Emergency Coolant Injection (ECI) system in providing inventory make-up to the heat transport system for a LOCA in the short term. Also, there is no single active or passive component failure that can result in failure of the Long Term Cooling (LTC) system in the long term (recirculation) after a LOCA. Therefore the ECC system (ECI and LTC) is designed to meet the single failure criterion.
In Reference [A2], AECL outlined a proposal for the treatment of limited core damage accidents (LCDAs) for consideration by the NRC staff. As noted in that communication, LOCA+LOECC is best considered separately from design basis accidents (on a low probability basis) and separately from severe core damage accidents (on a low consequence basis).

The events with loss of ECC shown in Table A4 (Table 2-2 of Reference [A5]) will be analysed assuming various credible failures of the ECC system. That is, the events considered should have a frequency within the LCDA range shown in Table A1.

References:
[A2] V. J. Langman, "Regulatory Treatment of Limited Core Damage Accidents for the ACR-700", AECL Letter to US NRC, May 10, 2004.
[A5] AECL Report 10810-03510-AB-001, "Initial Conditions and Standard Assumptions Safety Analysis Basis".

228. Please define the term, "limited failures, used in Table 2-3 of Reference 1 for the fuel acceptance criterion for Class 3 events.

Reference:

1. ACR-700 10810-03510-AB-001 (Rev. 0), "Initial Conditions and Standard; Assumptions Safety Analysis Basis", dated August 14, 2003.

AECL Response:
Limited fuel failures indicates that the extent of fuel cladding failures is limited such that the reference dose limits are met.

Table A1
Summary of Event Classification

Event Category	*Event Class*	*Typical Frequency Range (Events per year)*	*Effective dose Limit (Rem)*
DBE	*Class 1*	*>1.0E-02*	*0.05*
DBE	*Class 2*	*1.0E-02 – 1.0E-03*	*0.5*
DBE	*Class 3*	*1.0E-03 – 1.0E-04*	*3*
LCDA	*Class 4*	*1.0E-04 – 1.0E-05*	*10*
LCDA	*Class 5*	*1.0E-05 – 1.0E-06*	*25*
SCDA	-	*<1.0E-06*	-

Table A2
Examples of Design Basis Accidents

Design Basis Events	*Class*	*Grouped Event ID and Description*	*Preliminary Associated Frequency (event/year)*
Large LOCA	*3*	*GE-11 – Large LOCA*	*4.18E-04*
Large LOCA + LCIVP	*3*		*TBD*
Small LOCA	*2*	*GE-14 – Small LOCA (No containment by-pass)*	*7.07E-03*
Small LOCA + LCIVP	*3*		*TBD*
Pressurizer Pipe Break	*2*		*TBD*
Pressurizer Pipe Break + LCIVP	*3*		*TBD*
Single SG Tube Rupture	*2*	*GE-35 – Single SG tube rupture (leaks with containment by-pass)*	*8.04E-04*
Single SG Tube Rupture + LCIVP	*3*		*TBD*
Off-Stagnation Feeder Break	*2*	*GE-05 – Feeder break no flow stagnation*	*8.04E-04*
Off-Stagnation Feeder Break + LCIVP	*3*		*TBD*
Partial Single Channel Flow Blockage	*2*		*TBD*
Partial Single Channel Flow Blockage + LCIVP	*3*		*TBD*
PT Failure (CT intact)	*2*	*GE-15 - PT Failure (CT intact)*	*4.18E-03*
PT/CT Failure	*3*	*GE-13 – PT and CT rupture*	*4.18E-04*
PT/CT Failure + LCIVP	*3*		*TBD*
Loss of Reactivity Control	*1*		*TBD*
Total LCIVP	*1*	*GE-58 - Total loss of one unit class IV power supply*	*2.17E-01*
Partial LCIVP	*1*		*TBD*
Single HTS Pump Trip	*1*	*GE-21 – Partial loss of HTS pumped flow*	*2.89E-02*
Pump Seizure	*3*		*TBD*
Main Steam Line Break (inside containment)	*3*	*GE-41 - Main Steam Line Break (inside containment)*	*7.06E-05*
Main Steam Line Break (inside containment) + LCIVP	*3*		*TBD*

Feedwater Line Failure (inside containment)	*3*	*GE-38 – Asymmetric break inside reactor building downstream steam generator check valve*	*5.80E-05*
		GE-37 – Asymmetric break inside reactor building between steam generator check valve and containment wall	*1.70E-05*
Feedwater Line Failure (inside containment) + LCIVP	*3*		*TBD*

Table A3
Comparison Table of Event Categorization and Acceptance Criteria with SRP

Design Basis Events	*Class per Ref. [A5]*	*Revised Class*	*SRP Section Number*	*Accident Category*	*Key acceptance criteria from Ref. [A5]*	*Key acceptance criteria from SRP Ch. 15*	*Notes*
Large LOCA	*3*	*3*	*15.6.5*	*Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary*	• *Dose < 3 rem* • *No PT failure* • *H_2 concentration below flammability limit*	• *Dose < 25 rem* • *PCT < 2200 °F* • *H_2 < 1% hypothetical if all cladding reacted* • *Clad oxide thickness < 17%*	
Large LOCA + LCIVP	*3*	*3*	*15.6.5*	*Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary*	• *Dose < 3 rem* • *No PT failure* • *H_2 concentration below flammability limit*	• *Dose < 25 rem* • *PCT < 2200 °F* • *H_2 < 1% hypothetical if all cladding reacted* • *Clad oxide thickness < 17%*	*Consequential Loss of Class IV Power*
Small LOCA	*2*	*2*	*15.6.5*	*Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary*	• *Dose < 0.5 rem* • *No calculated clad failures* • *No PT failure*	• *Dose < 25 rem* • *PCT < 2200 °F* • *H_2 < 1% hypothetical if all cladding reacted* • *Clad oxide thickness < 17%*	
Small LOCA + LCIVP	*3*	*2*	*15.6.5*	*Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within*	• *Dose < 0.5 rem* • *No calculated clad failures*	• *Dose < 25 rem* • *PCT < 2200 °F* • *H_2 < 1%*	*Consequential Loss of Class IV Power*

Design Basis Events	Class per Ref. [A5]	Revised Class	SRP Section Number	Accident Category	Key acceptance criteria from Ref. [A5]	Key acceptance criteria from SRP Ch. 15	Notes
				the Reactor Coolant Pressure Boundary	• No PT failure	hypothetical if all cladding reacted • Clad oxide thickness < 17%	
Pressurizer Pipe Break	2	2	15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	• Dose < 0.5 rem • No calculated clad failures • No PT failure	• Dose < 25 rem • PCT < 2200 °F • H_2 < 1% hypothetical if all cladding reacted • Clad oxide thickness < 17%	Special case of Small LOCA because pressurizer low level trip is masked
Pressurizer Pipe Break + LCIVP	3	2	15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	• Dose < 0.5 rem • No calculated clad failures • No PT failure	• Dose < 25 rem • PCT < 2200 °F • H_2 < 1% hypothetical if all cladding reacted • Clad oxide thickness < 17%	Special case of Small LOCA because pressurizer low level trip is masked
Single SG Tube Rupture	2	2	15.6.3	Radiological Consequences of Steam Generator Tube Failure	Dose < 0.5 rem	Dose < 2.5 rem	
Single SG Tube Rupture + LCIVP	3	2	15.6.3	Radiological Consequences of Steam Generator Tube Failure	Dose < 0.5 rem	Dose < 2.5 rem	
Off-Stagnation Feeder Break	2	2	15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	• Dose < 0.5 rem • No calculated clad failures in unaffected channels • No PT failure in	• Dose < 25 rem • PCT < 2200 °F • H_2 < 1% hypothetical if all cladding reacted	Special case of Small LOCA – single channel event

Design Basis Events	Class per Ref. [A5]	Revised Class	SRP Section Number	Accident Category	Key acceptance criteria from Ref. [A5]	Key acceptance criteria from SRP Ch. 15	Notes
					unaffected channels	• Clad oxide thickness < 17%	
Off-Stagnation Feeder Break + LCIVP	3	2	15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	• Dose < 0.5 rem • No calculated clad failures in unaffected channels • No PT failure in unaffected channels	• Dose < 25 rem • PCT < 2200 °F • H_2 < 1% hypothetical if all cladding reacted • Clad oxide thickness < 17%	Special case of Small LOCA – single channel event
Partial Single Channel Flow Blockage	2	2	15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	• Dose < 0.5 rem • No calculated clad failures in unaffected channels • No PT failure in unaffected channels	• Dose < 25 rem • PCT < 2200 °F • H_2 < 1% hypothetical if all cladding reacted • Clad oxide thickness < 17%	Special case of Small LOCA – single channel event
Partial Single Channel Flow Blockage + LCIVP	3	2	15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	• Dose < 0.5 rem • No calculated clad failures in unaffected channels • No PT failure in unaffected channels	• Dose < 25 rem • PCT < 2200 °F • H_2 < 1% hypothetical if all cladding reacted • Clad oxide thickness < 17%	Special case of Small LOCA – single channel event
PT Failure (CT intact)	2	2	15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure	• Dose < 0.5 rem • No calculated clad failures in unaffected channels	• Dose < 25 rem • PCT < 2200 °F • H_2 < 1% hypothetical if all	Special case of Small LOCA – single channel event

Design Basis Events	Class per Ref. [A5]	Revised Class	SRP Section Number	Accident Category	Key acceptance criteria from Ref. [A5]	Key acceptance criteria from SRP Ch. 15	Notes
				Boundary	• No PT failure in unaffected channels	cladding reacted • Clad oxide thickness < 17%	
PT/CT Failure	3	3	15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	• Dose < 3 rem • No calculated clad failures in unaffected channels • No PT failure in unaffected channels	• Dose < 25 rem • PCT < 2200 °F • H_2 < 1% hypothetical if all cladding reacted • Clad oxide thickness < 17%	Special case of Small LOCA – single channel event
PT/CT Failure + LCIVP	3	3	15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	• Dose < 3 rem • No calculated clad failures in unaffected channels • No PT failure in unaffected channels	• Dose < 25 rem • PCT < 2200 °F • H_2 < 1% hypothetical if all cladding reacted • Clad oxide thickness < 17%	Special case of Small LOCA – single channel event
Loss of Reactivity Control	1	1	15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	• No fuel dryout for first reactor trip • Overpressure protection requirements met for RCS • No fuel failures	• No fuel dryout • Overpressure protection requirements met for RCS • No fuel C/L melting	Analysis is performed as a parametric analysis on reactivity insertion rate. Range of parametric study covers all credible means of reactivity insertion. Range will be justified in the
			15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power			
			15.4.3	Control Rod Misoperation (System Malfunction or Operator)			

Design Basis Events	Class per Ref. [A5]	Revised Class	SRP Section Number	Accident Category	Key acceptance criteria from Ref. [A5]	Key acceptance criteria from SRP Ch. 15	Notes
			15.4.9	Spectrum of Rod Drop Accidents (BWR)			analysis report. Range of initial powers from shutdown to full power.
			15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)			
Total LCIVP Partial LCIVP Single HTS Pump Trip	1	1	15.3.1 15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	• No fuel dryout for first reactor trip • Overpressure protection requirements met for RCS	• No fuel dryout • Overpressure protection requirements met for RCS	Flow controller malfunctions not applicable. Range of initial powers analyzed.
Pump Seizure	3	3	15.3.3 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	Overpressure protection requirements met for RCS	Overpressure protection requirements met for RCS	
Main Steam Line Break (inside containment)	3	3	15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)	Containment overpressure protection requirements met	Containment overpressure protection requirements met	
Main Steam Line Break (inside containment) + LCIVP	3	3	15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)	Containment overpressure protection requirements met	Containment overpressure protection requirements met	
Feedwater Line Failure (inside containment)	3	3	15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	• Dose < 3 rem • Overpressure	• Dose • Overpressure	

Design Basis Events	*Class per Ref. [A5]*	*Revised Class*	*SRP Section Number*	*Accident Category*	*Key acceptance criteria from Ref. [A5]*	*Key acceptance criteria from SRP Ch. 15*	*Notes*
					protection requirements met for RCS	*protection requirements met for RCS*	
Feedwater Line Failure (inside containment) + LCIVP	*3*	*3*	*15.2.8*	*Feedwater System Pipe Breaks Inside and Outside Containment*	• *Dose < 3 rem* • *Overpressure protection requirements met for RCS*	• *Dose* • *Overpressure protection requirements met for RCS*	

Table A4 Examples of Limited Core Damage Accidents

Limited Core Damage Accidents	*Class*
Large LOCA + LOECC	*5*
Small LOCA + LOECC	*5*
Pressurizer Pipe Break + LOECC	*5*
Single SG Tube Rupture + LOECC	*5*
Multiple SG Tube Failure	*5*
Off-Stagnation Feeder Break + LOECC	*5*
Stagnation Feeder Break	*5*
Partial Single Channel Flow Blockage + LOECC	*5*
Severe Flow Blockage	*5*
End Fitting Failure + LOECC	*5*
Main Steam Line Break (inside containment) + LOECC	*5*
Feedwater Line Failure (inside containment) + LOECC	*5*