Rates of Initiating Events at

U.S. Nuclear Power Plants

1988–2002

1 INITIAL PLANT FAULT AND FUNCTIONAL IMPACT CATEGORY DEFINITIONS

This section presents the definitions and rules used by the analysts to sort the LERs into the initial plant fault and functional impact categories. These definitions are applicable for both initial plant fault and functional impact categories. Examples are included in a given category to further explain the category use.

Unless stated otherwise, the categories listed below are applicable to both boiling water reactors (BWRs) and pressurized water reactors (PWRs).

Events defined by these categories must be associated with a manual or automatic reactor trip. The reactor trip must occur with the plant critical at or above the point of adding heat. The event must occur shortly before or shortly after the reactor trip. The event may contribute to the reactor trip (as a functional impact and/or initial plant fault) or may occur subsequent to the reactor trip (as a functional impact). Engineering judgment was applied to determine whether a failure event that occurs tens of minutes after the reactor trip should be classified under a functional impact category.

1.1 Definitions

- **Functional Impact Group Definition**

Definition. The event categories in the functional impact group includes risk-significant events that could impact the ability to remove decay heat. The functional impact group contains 26 categories under 12 headings. Event categories classified as general transients were excluded in the functional impact group. General transients are a compilation of all reactor trip events that had no direct impact on mitigating systems' ability to remove decay heat.

Purpose. The purpose of the functional impact group is to determine the frequency at which risk-significant events are likely to occur in association with the reactor trip, regardless of their order in the reactor trip sequence. The results presented in the main body of the report are focused on the analysis and evaluation of risk-important event categories from the functional impact group. This focus was chosen by the NRC to support several risk-informed regulatory activities. Frequency estimates of functional impact categories are best suited for PRA analyses where the occurrence of a risk-significant event category (e.g., total loss of the main feedwater system or steam generator tube rupture) is not specifically modeled in the accident sequence event tree as a conditional failure. For this case, the frequency of a functional impact category (or groups of similar categories) is used as the initiating event frequency for quantification of the appropriate event tree.

Event classification. For each reactor trip, the analysts examined the sequence of events occurring any time before and shortly after the reactor trip. Each occurrence of an event from the table of functional impact categories was noted in the database for each reactor trip event. One or more functional impact events may be identified in a single reactor trip event sequence. However, a reactor trip sequence may have no functional impact events as would be expected for most reactor trips.

For example, consider the case where a total loss of feedwater flow causes a plant transient resulting in a reactor trip and turbine trip, and then the loss of offsite power (due to, for example, the failure to transfer the plant electrical power source from the main generator to the preferred offsite power source). The functional impact categories applicable for this reactor trip sequence are Loss of Offsite Power (category B1) and Total Loss of Feedwater Flow (category P1). The order in which the functional impact events occur is not considered in this study. The turbine trip event was not selected, since the event was not a functional impact category.

As discussed in the example above, a reactor trip sequence may have multiple occurrences of functional impact events. About 9% of all functional impact events are multiple occurrences.

1.1.1 Initial Plant Fault Group

Definition. The initial plant fault is the first event in a sequence of events causing or leading to an unplanned, automatic, or manual reactor trip. The initial plant fault group contains 48 mutually exclusive categories under 13 headings. Twelve headings include risk-significant categories that could impact the ability to remove decay heat (e.g., loss of offsite power, loss-of-coolant accident, and total loss of condenser heat sink). These 12 headings and associated categories are identical to all of the risk-significant headings and categories used in the functional impact group. The initial plant fault group also includes an additional heading with 22 categories typically classified as general transients in PRAs. As described above, general transients are a compilation of all reactor trip initiators that had no direct impact on mitigating systems ability to remove decay heat.

The headings and categories associated with the initial plant fault group are listed in Table 2 and defined in Appendix A. Table B in Appendix B provides a cross-reference of the categories used in this report with the categories used in previous reports.

Purpose. The events in the initial plant fault group are used in the engineering analysis section of the main report to develop insights from the conditional occurrences of risk-significant events. The events from the initial plant fault and functional impact groups were merged to compare the number of risk-significant events occurring after the reactor trip initiator (i.e., initial plant fault event).

Frequency estimates of initial plant fault categories are best suited for PRA analyses where the occurrence of one or more risk-significant event categories (e.g., total loss of the main feedwater system or steam generator tube rupture) are specifically included in the accident sequence event tree model as a conditional failure. The combination of these conditional functional and/or system successes and failures are depicted along the top heading across the event tree. For this type of event tree model, the frequency of a initial plant fault category (or a group of similar categories) is used as the initiating event frequency for quantification of the event tree. The conditional probability of a risk-significant event category subsequent to the initial plant fault event can be estimated from the data in the appropriate initial plant fault and functional impact categories. However, if a particular event category is not modeled in the event tree as a conditional event, then the frequency estimate of the functional impact category, which includes all occurrences of the event in the frequency estimate, may be more appropriate as the initiating event frequency for event tree quantification.

Event classification. For each reactor trip event, the analysts examined the sequence of events leading to the reactor trip and selected the event that occurred first from the list of 48 initial plant fault categories. Only one initial plant fault category was selected for each reactor trip. For example, consider the previous case where a total loss of feedwater flow causes a plant transient that results in a reactor trip and turbine trip, and then the loss of offsite power. The initial plant fault category appropriate for this reactor trip sequence would be Total Loss of Feedwater Flow (category P1), since it happened first. In this example, the total loss of feedwater may be the result of the failure or misoperation of components in the main feedwater system or associated with another system. However, if the root cause could not be matched to a category from the initial plant fault group, then the next event in the reactor trip sequence that could be matched was selected.

1.1.2 Special Interest Group

A third group, designated as special interest, includes additional events that are often of interest but are not associated with an initial plant fault or functional impact category, such as diesel starts and loads, and a manual reactor trip that occurs after the event initiator (i.e., initial plant fault). Station blackout and Anticipated Transient Without Scram (ATWS) events fall into this category because these events involve an initiating event and a failure of a mitigating system. All applicable special interest events are flagged regardless of their place in the sequence of events. This information was collected for future studies.

• A. (Reserved)

• B. Loss of Offsite Power

(B1) Loss of Offsite Power (LOSP)

A simultaneous loss of electrical power to all safety-related buses that causes emergency power generators to start and supply power to the safety-related buses.

The offsite power boundary extends from the offsite electrical power grid to the output breaker (inclusive) of the stepdown transformer that feeds the first safety-related bus with an emergency power generator. The plant switchyard and service-type transformers are included within the offsite power boundary.

This category includes the momentary or prolonged degradation of grid voltage that causes all emergency power generators to start (if operable) and load onto its associated safety-related buses (if available).

This category does not include an LOSP event that occurs while the plant is shutdown. In addition, it does not include any momentary undervoltage event that results in the automatic start of all emergency power generators, but in which the generators do not tie on to their respective buses due to the short duration of the undervoltage.

• C. Loss of Safety-Related Bus

- **(C1) Loss of Vital Medium Voltage ac Bus ($\geq$600 V, <10 kV)**
- **(C2) Loss of Vital Low Voltage ac Bus (<600 V)**
- **(C3) Loss of Vital dc Bus**

Loss of a safety-related electrical bus is any sustained de-energization of a safety-related bus due to the inability to connect to any of the normal or alternative electrical power supplies. The bus must be damaged or its power source unavailable for reasons beyond an open, remotely-operated feeder-breaker from a live power source.

Examples include: supply cable grounds; failed insulators; damaged disconnects; transformer deluge actuations; or improper uses of grounding devices.

This category does not include a momentary de-energization of a bus caused by a slow automatic transfer to an available power source. Losses of all lower voltage buses caused by the loss of the medium voltage feeder bus are not classified under this category unless a lower voltage bus was damaged beyond use. A loss of power to a single component in another system because of a failed or mis-positioned breaker that does not affect the entire bus is not included in this category, but is instead classified as a failure of the system that the single breaker serves. For example: a circuit breaker failure that causes a loss of power to a condensate pump that results in inadequate condensate flow, would be classified as a Partial Loss of Condensate (QP4).

• D. Loss of Instrument or Control Air System

(D1) Loss of Instrument or Control Air System

A total or partial loss of an instrument or control air system that leads to a reactor trip or occurs shortly after the reactor trip.

Examples include: ruptured air headers; damaged air compressors with insufficient backup capability; losses of power to air compressors; line fitting failures; improper system line-ups; and undesired operations of pneumatic devices in other systems caused by low air header pressure.

This category does not include a loss of air to a single component in another system because of a blockage or incorrect line-up that does not affect the header pressure, but is instead classified as a failure of the system that the single component serves. For example: a solenoid valve malfunction that causes a loss of plant air to a single feedwater valve and causes the feedwater valve to shut, would be classified as a Partial Loss of Feedwater (QP2). A loss of a redundant component in the air system is not classified as a partial loss of the air system as long as the remaining, similar components provide the required level of performance.

• E. Loss of Safety-Related Cooling Water

• (E1) Total Loss of Service Water

• (E2) Partial Loss of Service Water

A service water system (SWS) can be an open-cycle or a closed-cycle cooling water system. An open-cycle SWS takes suction from the plant's ultimate heat sink (e.g., the ocean, bay, lake, pond or cooling towers), removes heat from safety-related systems and components, and discharges the water back to the ultimate heat sink. A closed-cycle or intermediate SWS removes heat from safety-related equipment and discharges the heat through a heat exchanger to an open-cycle service water system.

These categories include the total or partial loss of a safety-related SWS, or a non-safety-related SWS that provides cooling to safety-related components during normal plant operations. For the latter case, a standby safety-related service water system automatically starts upon the loss of the non-safety-related system or during an accident sequence initiation.

Partial Loss of Service Water is a loss of one train of a multiple train system or partial loss of a single train system that impairs the ability of the system to perform its function. Examples include pump cavitation; strainer fouling; and piping rupture.

These categories do not include a loss of a redundant component in a SWS as long as the remaining, similar components provide the required level of performance. For example, a loss of a single SWS pump is not classified as a partial loss of a SWS as long as the remaining operating or standby pumps can provide the required level of performance. A loss of service water to a single component in another system because of a blockage or incorrect line-up that does not affect the cooling to other components serviced by the train is not included under this category, but is instead classified as a failure of the system that the single component serves.

• F. Steam Generator Tube Rupture: PWR (SGTR)

A rupture of one or more steam generator tubes that result in a loss of primary coolant to the secondary side of the steam generator at a rate greater than or equal to 100 gpm.

A SGTR can occur as the initial plant fault, such as a tube rupture caused by high cycle fatigue or loose parts, or because of another initiating event. The latter case would be classified as a functional impact.

This category applies to PWRs only. This category includes excessive leakage caused by the failure of a previous SGTR repair (i.e., leakage past a plug).

• G. Loss-of-Coolant Accident (LOCA)/Leak

(G1) Very Small LOCA/Leak

A pipe break or component failure that results in a loss of primary coolant between 10 to 100 gpm, but does not require the automatic or manual actuation of high-pressure injection systems.

Examples include: reactor coolant pump (PWR) or recirculating pump (BWR) seal failures; valve packing failures; steam generator tube leaks; and instrument line fitting failures.

Note: Leakage from a pressurizer code safety valve (PWR), main steam line safety valve (BWR), or Automatic Depressurization System relief valve (BWR) are classified under category G2 or G5. Leakage from a pressurizer power operated relief valve is classified under category G4. A steam generator tube rupture (PWR) is classified under category F1. A small primary system leak (less than 10 gpm) that results in a manual reactor trip is classified under category Primary System Leak (QG9). Category QG9 only applies to the initial plant fault group, however.

(G2) Stuck Open: 1 Safety/Relief Valve

A failure of one primary system safety and/or relief valve (SRV) to fully close that results in the loss of primary coolant.

The valves included in this category are pressurizer code safety valves (PWR), main steam line safety valves (BWR), and Automatic Depressurization System relief valves (BWR). The stuck open SRV may or may not cause the automatic or manual actuation of high-pressure injection systems.

This category includes a stuck open valve that cannot be subsequently closed upon manual demand (BWRs) or does not subsequently close on its own immediately after the reactor trip (BWRs). The mechanism that opens the valve is not a defining factor. The different mechanisms than can open an SRV are transient-induced opening, manual opening during valve testing (BWRs), and spurious opening.

In BWRs, only a stuck open SRV event initiated by routine surveillance testing of the valve during power operations would be classified as an initial plant fault under this category since no other initial plant fault category applies. All stuck open single SRV events in BWRs and PWRs are classified as a functional impact under this category. An inadvertent open SRV event during power operations which closes on its own after the reactor trip and before the manual or automatic actuation of a high pressure injection system are classified under category QG10, Inadvertent Open/Close: 1 Safety/Relief Valve. Category QG10 only applies to the initial plant fault group, however.

This category does not include a weeping safety valve.

Note: A stuck open pressurizer power-operated relief valve (PWR) is classified under category G4.

(G3) Small Pipe Break LOCA

For a BWR, a pipe in the primary system boundary with a break size less than 0.004 ft^2 (or a 1 inch inside diameter pipe equivalent) for liquid and less than 0.05 ft^2 (or an approximately 4 inch inside diameter pipe equivalent) for steam. For a PWR, a pipe break in the primary system boundary with an inside diameter between ½ to 2 inches.

The above break size ranges were used in the NUREG150 analyses of a selected group of plants. The plant-specific range of LOCA sizes should be divided into groups for which plant response, in terms of required system operability, is the same or very similar. The following generic definition was used in NUREG150: a small break LOCA is a break that does not depressurize the reactor quickly enough for the low pressure systems to automatically inject and provide sufficient core cooling to prevent core damage. However, low capability systems (i.e., 100 to 1500 gpm) are sufficient to make up the inventory completion.

Note: A steam generator tube rupture is classified under category F1. A steam generator tube leak is classified under category G1 or QG9. A stuck open safety or relief valve is classified under category G2, G4, or G5.

(G4) Stuck Open: Pressurizer PORV

A pressurizer power-operated relief valve (PORV) that fails to close.

This category applies to PWRs only.

(G5) Stuck Open: 2 or More Safety/Relief Valves

Two or more primary system safety and/or relief valves that fails to close.

The valves included in this category are pressurizer code safety valves (PWR), main steam line safety valves (BWR), and Automatic Depressurization System relief valves (BWR).

This category does not include a weeping safety valve.

Note: A stuck open pressurizer power-operated relief valve (PWR) is classified under category G4, Stuck Open PORV.

- **(G6) Medium Pipe Break LOCA**

For a BWR, a pipe in the primary system boundary with a break size between 0.004 to 0.1 ft^2 (or an approximately 1 to 5 inches inside diameter pipe equivalent) for liquid and between 0.05 to 0.1 ft^2 (or an approximately 4 to 5 inches inside diameter pipe equivalent) for steam. For a PWR, a pipe break in the primary system boundary with an inside diameter between 2 to 6 inches.

The above break size ranges were used in the NUREG150 analyses of a selected group of plants. The plant-specific range of LOCA sizes should be divided into groups for which plant response, in terms of required system operability, is the same or very similar. The following generic definition was used in NUREG150: a medium break LOCA is a break that does not depressurize the reactor quickly enough for the low pressure systems to automatically inject and provide sufficient core cooling to prevent core damage. However, the loss from the break is such that high capacity systems (i.e., 1500 to 5000 gpm) are needed to makeup the inventory depletion.

- **(G7) Large Pipe Break LOCA**

For a BWR, a pipe in the primary system boundary with a break size greater than 0.1 ft^2 (or an approximately 5 inches inside diameter pipe equivalent) for liquid and steam. For a PWR, a pipe break in the primary system boundary with an inside diameter greater than 6 inches.

The above break size ranges were used in the NUREG150 analyses of a selected group of plants. The plant-specific range of LOCA sizes should be divided into groups for which plant response, in terms of required system operability, is the same or very similar. The following generic definition was used in NUREG150: a large break LOCA is a break that depressurizes the reactor to the point where the low pressure systems can injection automatically providing sufficient core cooling to prevent core damage.

- **(G8) Reactor Coolant Pump Seal LOCA: PWR**

A catastrophic failure the reactor coolant pump seal assembly that results in a primary coolant leak into the primary containment at a rate greater than 100 gpm.

This category applies to PWRs only.

A reactor coolant pump seal leak with a leak rate less than 100 gpm is classified under category G1 or QG9.

• H. Fire

- **(H1) Fire**

Smoke or flames inside the plant or site boundary that results in damage to safety- or non-safety-related equipment.

Examples include: fires located in the plant switchyard (e.g., transformers, switchgear); burning thermal or electrical insulation; transformer, circuit breaker, and power supply fires; rags ignited by hot relief valve tailpipes; burning lube oil; and offsite brush fires that caused a loss of an electrical power transmission line. Fire-related events classified under this category typically require a response by plant personnel; however, damage to plant equipment determined in the post event evaluation to be caused by a fire that went undetected is also included in this category.

This category does not include a smoldering lightning arrestor caused by a lightning strike; the “smoking” of a set of breaker auxiliary contacts or a small relay coil; a simple fire in a trash can or ash tray; or a fire to an administrative support building (e.g., trailer) that does not effect plant structures, equipment or components required to maintain the plant in a safe condition.

• J. Flood

- **(J1) Flood**

A major on-site pipe break other than a high energy line break (as defined by heading K) that causes damage to structures, equipment, or components.

An example of this is leakage from condensate or feedwater lines (as defined under category QK4) as long as the leakage resulted in damage to structures, equipment, or components.

This category does not include an activation of a transformer deluge system or natural flooding (e.g., river overflowing, heavy rains, etc.).

• K. High Energy Line Break

- **(K1) Steam Line Break Outside Containment**

A break of one inch equivalent diameter or more in a steam line located outside the primary containment that contains main turbine working fluid at or above atmospheric saturation conditions.

Examples include: operation of rupture disks; and breeches of a pipe caused by a split, crack, weld failure, or circumferential break.

- **(K2) Feedwater Line Break**

A break of one inch equivalent diameter or more in a feedwater or condensate line that contains main turbine working fluid at or above atmospheric saturation conditions.

Examples include: breeches of a pipe caused by a split, crack, weld failure, or circumferential break.

- **(K3) Steam Line Break Inside Containment (PWR)**

A break of one inch equivalent diameter or more in a steam line located inside the primary containment that contains main turbine working fluid at or above atmospheric saturation conditions.

This category applies to PWRs only. Examples include: breeches of a pipe caused by a split, crack, weld failure, or circumferential break.

• L. Total Loss of Condenser Heat Sink

- **(L1) Inadvertent Closure of All MSIVs**

A complete closure of at least one MSIV in each main steam line.

An example includes the automatic closure of all MSIVs as part of an engineered safety feature actuation.

This category does not include a manual closure of all MSIVs to limit cool down rate after a reactor trip, as long as the MSIVs are capable of being reopened by operator demand.

- **(L2) Loss of Condenser Vacuum**

A decrease in condenser vacuum that leads to an automatic or manual reactor trip, or manual turbine trip; or a complete loss of condenser vacuum that prevents the condenser from removing decay heat after a reactor trip.

The main condenser boundary includes the condenser air ejectors and condenser vacuum pumps.

Initial plant faults that contribute to a loss of condenser vacuum include: circulating water pump trips (category QL4); traveling screen blockage (category QL4); and condenser leakage (category QL6).

This category does not include the loss of condenser vacuum caused by the loss of offsite power.

In addition, reactor trips that are the indirect result of a low condenser vacuum, such as a loss of feedwater caused by condensate pumps tripping on high condensate temperature because of loss of vacuum, are counted as Loss of Condenser Vacuum.

A loss of condenser vacuum resulting from a manual trip in response to a plant event that had no direct effect on the main condenser vacuum was not included in this category.

Note: In BWRs, a low condenser vacuum signal will generate a reactor trip. In PWRs, a low vacuum signal will cause turbine-driven main feedwater pumps to trip, which will result in a reactor trip on low steam generator level.

- **(L3) Turbine Bypass Unavailable**

The failure of one or more turbine bypass valves (TBVs) to maintain the reactor pressure and temperature at the desired operating condition.

Turbine bypass failures included in this category may result in an automatic or manual reactor trip during an unsuccessful turbine run back; and the sustained use of one or more atmospheric dump valves (PWR) or safety relief valves to the suppression pool (BWR) after the reactor trip.

This category does not include turbine bypass valve closures caused by the loss of offsite power.

M. (Reserved)

N. Interfacing System LOCA

- **(N1) Interfacing System LOCA**

A backflow of high-pressure coolant from the primary system through low pressure system piping which results in the breach of the pipe or component.

P. Total Loss of Feedwater Flow

- **(P1) Total Loss of Feedwater Flow**

A complete loss of all main feedwater flow.

Examples include: the trip of the only operating feedwater pump while operating at reduced power; the loss of a startup or an auxiliary feedwater pump normally used during plant startup; the loss of all operating feed pumps due to trips caused by low suction pressure, loss of seal water, or high water level (BWR reactor level or PWR steam generator level); anticipatory reactor trip due to loss of all operating feed pumps; and manual reactor trip in response to feed problems characteristic of a total loss of feedwater flow, but prior to automatic Reactor Protection System signals.

This category also includes the inadvertent isolation or closure of all feedwater control valves prior to the reactor trip, however, a main feedwater isolation caused by valid automatic system response after a reactor trip is not included.

This category does not include the total loss of feedwater caused by the loss of offsite power.

Q. General Transients

Categories under this heading are only used for initial plant faults, not for functional impact classification. The general transient categories result in automatic or manual reactor trips but do not degrade safety system response. Because these categories are only applicable as an initial plant fault, they will only be used when the event they describe is the first event from this entire list of categories to occur.

- **(QC4) Loss of ac Instrumentation and Control Bus**

A sustained de-energization of an ac instrumentation and control bus due to the inability to connect to any of the normal or alternative electrical power supplies.

An event classified in this category is normally associated with damage to the bus itself, or damage to its uninterruptable power supply or supply breaker. The bus had to be damaged or its power source unavailable for reasons other than a remotely-operated feeder-breaker being open. This category includes only those failures of safety- and non-safety-related ac instrumentation and control buses that lead to an automatic or manual reactor trip.

This category does not include a momentary undervoltage of a bus caused by a slow automatic transfer, or a loss of one output from the bus (e.g., failure of one output breaker), but is instead classified as a loss of the affected system.

- **(QC5) Loss of Non-safety-Related Bus**

A sustained deenergization of a non-safety-related bus other than an ac instrumentation and control bus due to the inability to connect to any of the normal or alternative electrical power supplies.

This category is normally associated with damage to the bus itself, or damage to its feeder transformer or supply breaker. The bus had to be damaged or its power source unavailable for reasons other than a remotely-operated feeder-breaker being open. This category includes faults to high (>10kV), medium (>600V, <10KV), and low (>120V, <600V) non-safety-related ac buses that lead to an automatic or manual reactor trip.

This category does not include a momentary undervoltage of a bus caused by a slow automatic transfer, or a loss of one output from the bus (e.g., failure of one output breaker), but is instead classified as a failure of the affected system.

There were no events classified as failures of a non-safety-related dc bus.

(QG9) Primary System Leak

A small leak of primary coolant, inside the primary containment, at a rate less than 10 gpm and results in an automatic or manual reactor trip. A crack in one or more steam generator tubes that result in a loss of primary coolant to the secondary side of the steam generator at a rate less than 10 gpm.

A plant shutdown is required for primary leak rates that exceeds technical specification limits. Most shutdown events of this nature do not result in a reactor trip. Manual reactor trips are sometimes initiated to expedite the controlled shutdown to avoid violating technical specification requirements. Automatic reactor trips may occur during the controlled shutdown caused by problems not related to the leak itself. Examples include feedwater flow problems at low power or operator errors.

This category includes those primary leak events that prompt a controlled reactor shutdown and somehow result in an automatic or manual reactor trip.

(QG10) Inadvertent Open/Close: 1 Safety/Relief Valve

One or more primary system safety and/or relief valves that inadvertently opens during normal power operations and then closes on its own prior to the manual or automatic actuation of a high pressure injection system.

The valves included in this category are pressurizer code safety valves (PWR), pressurizer power-operated relief valves (PWR), main steam line safety valves (BWR), and Automatic Depressurization System relief valve (BWR).

This category only applies to the initial plant fault group.

- **(QK4) Steam or Feed Leakage**

A loss of the main turbine working fluid at or above atmospheric saturation conditions from the steam or main feedwater system up to and including a pipe break less than one-inch equivalent diameter.

This category includes a small steam or feedwater leak that leads to an automatic or manual reactor trip. Examples include: flange leaks, packing leaks, blown fittings and leaks through other system connections.

Note: Pipe breaks one inch equivalent diameter or more are classified under heading K, High Energy Line Breaks.

- **(QL4) Loss of Non-safety-Related Cooling Water**

A total or partial loss of a non-safety-related cooling water system that leads to an automatic or manual reactor trip.

This category includes the loss of non-safety-related cooling water systems that provide cooling to non-safety-related balance-of-plant components. Examples of cooling water systems include turbine building service water systems, non-safety-related mechanical draft cooling towers, and condenser circulating water systems.

This category does not include partial or total loss of a safety-related cooling water system (e.g., service water, component cooling water) that results in a reactor trip due to the loss of non-safety-related balance-of-plant equipment in which it serves during normal plant operations. This event is classified under the appropriate category under heading E, Loss of Safety-Related Cooling Water.

- **(QL5) Partial Closure of MSIVs**

Any combination of partial or full closure of one or more main steam isolation valves (MSIVs) with at least one main steam line open to pass steam to the main condenser.

This category includes partial MSIV closures that lead to an automatic or manual reactor trip. Examples include: full closure of one MSIV and partial closure of one MSIV.

Note: In BWRs, a reactor trip signal will be generated by the closure of any single MSIV.

- **(QL6) Condenser Leakage**

Faults in the condenser shell, tubing, or connective components that result in leakage (fluid or gas) to or from the condenser.

Examples include condenser expansion joint ruptures or leaks, tube leaks/ruptures that require shutdown for conductivity/chemistry although condenser vacuum is normal, and breaks in piping attached to the condenser.

- **(QP2) Partial Loss of Feedwater Flow**

A reduction in main feedwater flow that leads to an automatic or manual reactor trip.

All main feedwater system component malfunctions in conjunction with a steam generator low level alarm were considered to be at least a partial loss of feedwater (if not a total loss of feedwater as defined by category P1). Examples include the partial or full closure of a feedwater regulation valve, and a trip of one feedwater pump.

This category does not include steam generator level shrinkage events due to the injection of colder water (usually during low power operations). In addition, protective trip of a single main feedwater pump due to inadequate suction pressure caused by a partial loss of condensate flow is classified under QP4, Partial Loss of Condensate Flow.

- **(QP3) Loss of Condensate Flow**

A complete loss of condensate flow that leads to an automatic or manual reactor trip.

Examples include: the failure of all condensate pumps or booster pumps; and a malfunction that causes a loss of all condensate flow to the main feedwater system.

Note: An event that results in a total loss of condensate flow as the initial plant fault will result in the total loss of feedwater flow, therefore, this event will also be classified under function impact category P1, Total Loss of Feedwater Flow.

- **(QP4) Partial Loss of Condensate Flow**

A reduction in condensate flow that leads to an automatic or manual reactor trip.

Examples include: the failure of less than all condensate pumps; and a fault in the feed heater or condensate path that causes a reduction of condensate flow.

- **(QP5) Excessive Feedwater Flow**

An inadvertent increase in feedwater flow that leads to an automatic or manual reactor trip.

Excessive feedwater transients as the initiating event can generate various reactor protection system (RPS) trip signals. Examples of events caused by excessive feedwater as the transient initiator include: an automatic reactor trip on high rate of power change caused by moderator temperature effects (BWR); a turbine trip/reactor trip due to high reactor water (BWR) or high steam generator (PWR) level; a reactor water (BWR) or steam generator (PWR) low-level reactor trip that follow the tripping of feedwater pumps caused by high levels due to excessive feedwater flow; and a manual reactor trip in response to improper feedwater regulation valve operation.

This category does not include a transient that results in a swell (increased level) in the reactor vessel or steam generator caused by other than excessive feedwater flow (usually depressurization of steam).

- **(QR0) RCS High Pressure (RPS Trip)**

A transient not classified under any other category that causes reactor pressure to increase to the high pressure RPS trip setpoint.

- **(QR1) RCS Low Pressure (RPS Trip): PWR**

A transient not classified under any other category that causes primary pressure to decrease to the low pressure RPS trip setpoint.

This category only applies to PWRs.

- **(QR2) Loss of Primary Flow (RPS Trip): PWR**

A total loss or reduction in reactor coolant system flow that results in a RPS trip.

This category only applies to PWRs. Examples of events that may cause a reactor coolant pump (RCP) trip include momentary undervoltage transients and RCP faults.

This category does include RCP trips caused by a damaged RCP electrical bus as a result of Fire (H1).

- **(QR3) Reactivity Control Imbalance**

A reactivity anomaly that leads to an automatic or manual reactor trip.

Examples include: high negative or positive neutron flux rate RPS trip (PWR); and automatic and manual reactor trips caused by a dropped control rod, an inadvertent rod withdrawal, a rod control system malfunction, a neutron flux imbalance, or an indication of core instability.

This category does not include a reactivity anomaly that results in a high reactor power RPS trip (classified under category QR4, Core Power Excursion.)

- **(QR4) Core Power Excursion (RPS Trip)**

A reactivity anomaly that causes reactor power exceeding the high reactor power RPS trip setpoint.

Examples of events that typically cause a high reactor power RPS trip include: an inadvertent rod withdrawal that do not cause a high neutron flux rate RPS trip; improper operation of the mode selector switch during startup or shutdown that enables a power level trip lower than the present power level (BWR); a neutron flux spike due to pressure changes or recirculation flow changes (BWR); steam pressure oscillations caused by a turbine control system malfunction (BWR); neutron flux exceeding the flow-biased average power range monitor (APRM) scram setpoint (BWR); and a power increase caused by overfeeding cold feedwater (BWR).

- **(QR5) Turbine Trip**

An inadvertent trip of the main turbine that results in a cessation of steam flow to the turbine, and leads to an automatic or manual reactor trip.

Manual turbine trips performed to cause a reactor trip were never an initiating event. Events of this type were considered as special interest group.

The main turbine as defined in this category includes the main turbine and its auxiliaries; the electrohydraulic control system; turbine throttle valves; main generator and its auxiliaries; and the main generator output breakers.

This category includes: inadvertent closure of all turbine throttle valves; EHC fault; main generator trip due to a switchyard equipment fault (e.g., output breaker, main transformer, switchyard breaker, offsite transmission line); response to electrical grid undervoltage voltage or frequency anomaly; inadequate plant response to an electric load rejection; unplanned manual turbine trip; and a spurious turbine trip.

- **(QR6) Manual Reactor Trip**

A manual initiation of a reactor trip, either purposely or by human error.

This category does not include: the improper operation of the mode selector switch during startup or shutdown that enables a power level trip lower than the present power level (classified under category QR4, Core Power Excursion).

- **(QR7) Other Reactor Trip (Valid RPS Trip)**

All other reactor trips (other than those listed above) that result when an actual plant condition reaches the RPS trip setpoint for that condition.

- **(QR8) Spurious Reactor Trip**

An automatic reactor trip caused by hardware failure or human error in a RPS instrumentation or logic channel, or a reactor trip breaker.

Examples include: incorrect venting of an instrument line during maintenance that causes false signal being sent to the RPS; and any other RPS system fault or human error that generates a reactor trip signal that does not reflect actual plant conditions.

- **(QR9) Spurious Engineered Safety Feature Actuation**

A spurious actuation of the Engineering Safety Features (ESF) system caused by hardware failure or human error in an ESF instrumentation or logic channel that results in a reactor trip.