15.8 Anticipated Transients Without Scram

15.8.1 General Background

10 CFR 50.62 defines an anticipated transient without scram (ATWS) as an anticipated operational occurrence (AOO) followed by failure of the reactor trip (RT) portion of the protection system (PS). 10 CFR 50, Appendix A defines AOOs as those conditions during normal operation that are expected to occur one or more times during the life of the nuclear power unit. These conditions include but are not limited to loss of power to the recirculation pumps (boiling water reactors only), tripping of the turbine generator set, isolation of the main condenser, and loss of offsite power.

10 CFR 50.62(c)(1) requires that "each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the RT system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing RT system."

Design features related to diverse scram signals have also been included in the U.S. EPR design.

15.8.1.1 U.S. EPR Design Features

The concept of defense-in-depth is used in the U.S. EPR by establishing and protecting barriers to fission product release, such as fuel cladding, reactor coolant system (RCS) boundary, and containment. These barriers are protected by an appropriate set of design provisions and operating strategies, which include but are not limited to the following:

- Classification of functions, systems, and structures according to their importance to safety.
- Deterministic design requirements according to safety class regarding redundancy, diversity, and material quality.
- Probabilistic design objectives balancing frequency of occurrence and potential consequences.
- Conservative design.

In addition, design provisions and operating strategies are staggered according to lines of defense that:

- Prevent departures from normal operation by surveillance and control systems.

- Implement interlocks and preventive actions that cope with deviations from normal operation to prevent an event resulting in the actuation of a safety system.
- Mitigate events and bring the plant to a controlled state by the protection system that controls the RT actuation system and the engineered safety features.
- Cope with the complete failure of engineered safety features and protective systems by means of risk reduction functions.
- Preserve containment integrity in case of accidents resulting in mass and energy releases into the containment.

The following sections address U.S. EPR design features that function to mitigate or reduce the risk of ATWS events.

15.8.1.2 Protection System

The PS provides the primary means for protecting fission product barriers by tripping the reactor. A description of the PS is provided in Section 7.2.

An ATWS event occurs when the control rods fail to insert following the generation of an RT signal. The specific failure mechanism is not specified, but could be the result of mechanical blockage of the control rods or the result of electrical or mechanical failures within the PS. If the ATWS is the result of a mechanical blockage of the control rods, failures within the PS are not postulated. Details of mechanical blockage are provided in Section 15.8.1.6.

If the ATWS is the result of a failure within the PS, an independent and diverse RT signal within the process automation system (PAS) bypasses the PS and initiates an RT. The RT is initiated by opening the breakers mounted at the output of the motor-generator sets that energize the control rod drive mechanisms. The diverse system also processes other critical signals and initiates essential actions to trip the turbine and start other safeguards systems as needed to address the ATWS event. Details of these signals are addressed in Section 15.8.1.3.

15.8.1.3 Process Automation System

The main tasks performed by the PAS are monitoring and automation of plant mechanical and electrical systems for normal operating conditions and postaccident conditions. The PAS also includes logic that fulfills the ATWS requirements of 10 CFR 50.62. The PAS logic is independent from sensor output to the final actuation device from the PS design features, and provides a diverse means to trip the reactor, trip the turbine, and initiate emergency feedwater (EFW) on conditions indicative of an ATWS. These diverse functions provided by the PAS provide reasonable assurance that a pressure increase does not exceed the ASME Service Level C limit of 3200 psig (Reference 1) or does not exceed containment safety parameters.

The diverse trip functions and capabilities are incorporated within PAS and used for ATWS mitigation are described in Section 7.8.1.2.

15.8.1.4 Emergency Feedwater System

The U.S. EPR provides automatic actuation of the EFW system on conditions indicative of an ATWS (see Section 15.8.1.3). The U.S. EPR is designed so that flow from the EFW system is not required for the first 30 minutes following an ATWS.

15.8.1.5 Extra Borating System

The extra borating system (EBS) is not required for ATWS mitigation. However, the system is available via manual actuation should additional negative reactivity be desired to bring the reactor into a subcritical state. A description of the EBS is provided in Section 6.8.

15.8.1.6 Mechanical Blockage of Rod Cluster Control Assemblies

ATWS events resulting from mechanical blockage of control rods are not postulated for the U.S. EPR design. The probability of an ATWS resulting from mechanical blockage of the control rods is an insignificant contributor to the overall probability of an ATWS.

As noted in the “Conclusions” section of NUREG-1780 (Reference 2), during ATWS rulemaking the NRC staff set a goal that the probability of an ATWS should be no more than 1.0E-05 per reactor year. The probability of an ATWS is defined by NUREG-1780 as “the annual frequency of an ATWS leading to plant conditions that exceed certain design parameters that can result in core melt, containment failure, and the release of radioactivity and can be viewed as the expected CDF of an unmitigated ATWS.” NUREG-1780 updated the original generic ATWS regulatory analysis using operating data since the ATWS rule was implemented, and the updated results indicated that the four reactor types achieved the ATWS rule risk goal: General Electric, Westinghouse, Babcock and Wilcox (B&W), and Combustion Engineering (CE). Specifically, Table 3, Summary of ATWS Rule Risk Expectations and Outcomes, of NUREG-1780 shows that the pressurized water reactor vendors (Westinghouse, B&W, and CE) each achieved a probability of an ATWS that is at least a factor of 20 better than the goal (i.e., $\leq$5.0E-07 per reactor year).

Inherent to the development of ATWS probability is the assumption of successful mitigation upon insertion of 20 percent or more of the control rods per SECY-83-293 (Reference 3). The ATWS rule 10 CFR 50.62(c)(2) requires that reactors designed by CE and B&W have a diverse scram system. The diverse scram system increases the reliability for control rod insertion, which is reflected in the probability of having a common-cause failure that causes 50 percent or more of the control rods to fail insertion. These probability values include 3.6E-08 for CE per Table 3-2 of NUREG/

CR-5500, Vol. 10 (Reference 4) and 4.1E-08 for B&W per Table 3-2 of NUREG/CR-5500, Vol. 11 (Reference 5). The U.S. EPR diverse scram system, as described in Section 15.8.1.3, significantly reduces the probability of having a common-cause failure for 50 percent or more of the control rods to fail. These probability values include the contribution from mechanical blockage. Therefore, the probability of an ATWS resulting from mechanical blockage of the control rods is an insignificant contributor to the overall probability of an ATWS.

15.8.2 Anticipated Transients Without Scram

15.8.2.1 Loss of Feedwater

The immediate consequence of a loss of main feedwater (MFW) flow is a reduction in the SG water level. If not corrected, the water level decrease ultimately results in an RT and EFW actuation. The loss of MFW flow causes the temperature of the SG water to increase, which causes the reactor coolant temperature and pressure to increase. Both safety systems, such as EFW, main steam relief trains (MSRT), and main steam safety valves (MSSV) and non-safety (turbine bypass) systems are available to remove sensible and decay heat. The PS provides the primary RT via the low SG level (MIN1) function to prevent a temperature increase that could cause fuel damage or a pressure increase that would challenge the integrity of the reactor coolant pressure boundary (RCPB). Also, EFW is actuated on a low SG level (MIN2) signal.

The U.S. EPR maintains the integrity of fission product barriers in the event of a PS failure by providing a diverse means to trip the reactor using a low SG narrow range level function. The U.S. EPR also provides a diverse means to trip the turbine following the diverse RT and a diverse means to initiate EFW on low SG level.

15.8.2.2 Loss of Electrical Load

In a loss of electrical load event, an electrical disturbance causes the loss of a significant portion of the generator load. Offsite AC power remains available to operate the station auxiliaries (e.g., reactor coolant pumps), therefore, emergency diesel generators are not required. When a loss of generator load occurs, immediate fast closure of the turbine control valves (TCV) and the intercept valves are initiated. The sudden reduction in steam flow causes the pressure and temperature in the secondary side of the SG to increase. As a result, there is an increase in reactor coolant temperature, a decrease in coolant density, an increase in water volume in the pressurizer, and an increase in reactor coolant pressure. Both safety (EFW, MSRTs, and MSSVs) and non-safety (turbine bypass) systems are available to remove sensible and decay heat. The PS provides the primary RT via the high SG pressure (MAX1) or high pressurizer pressure (MAX2) functions to prevent a pressure increase that would challenge the integrity of the RCPB. Also, EFW is actuated on a low SG level (MIN2) signal.

The U.S. EPR maintains the integrity of fission product barriers in the event of a PS failure by providing a diverse means to trip the reactor using either a high SG pressure function or a high pressurizer function. The U.S. EPR also provides a diverse means to trip the turbine following the diverse RT and a diverse means to initiate EFW on low SG level.

15.8.2.3 Turbine Trip

In a turbine trip event, a malfunction of a turbine or reactor system causes the turbine to trip offline by stopping steam flow to the turbine. Offsite AC power remains available to operate the station auxiliaries such as reactor coolant pumps, therefore, emergency diesel generators are not required. The turbine stop valves (TSV) are closed to interrupt steam flow to the turbine. These valves close faster than the TCVs, thereby producing a more severe transient than a loss of electrical load event. The sudden reduction in steam flow causes the pressure and temperature in the secondary side of the SG to increase. As a result, there is an increase in reactor coolant temperature, a decrease in coolant density, an increase in water volume in the pressurizer, and an increase in reactor coolant pressure. Both safety (EFW, MSRTs, and MSSVs) and non-safety (turbine bypass) systems are available to remove sensible and decay heat. The PS provides the primary RT via the high SG pressure (MAX1) or high pressurizer pressure (MAX2) functions to prevent a pressure increase that would challenge the integrity of the RCPB. Also, EFW is actuated on a low SG level (MIN2) signal.

The U.S. EPR maintains the integrity of fission product barriers in the event of a PS failure by providing a diverse means to trip the reactor using either a high SG pressure function or a high pressurizer pressure function. The U.S. EPR also provides a diverse means to trip the turbine following the diverse RT and a diverse means to initiate EFW on low SG level.

15.8.2.4 Loss of Condenser Vacuum

The loss of condenser vacuum event is one of the malfunctions that can cause a turbine trip. The event is analyzed as a turbine trip with a simultaneous loss of feedwater to the SGs due to low suction pressure on the feedwater pumps. Offsite AC power remains available to operate the station auxiliaries. The sudden reduction in steam flow resulting from closure of the TSVs coupled with the loss of cooling water causes the pressure and temperature in the secondary side of the SG to rapidly increase. As a result, the reactor coolant temperature increases, the coolant density decreases, the water volume in the pressurizer increases, and the reactor coolant pressure increases. Both safety (EFW, MSRTs, and MSSVs) and non-safety (turbine bypass) systems are available to remove sensible and decay heat. The PS provides the primary RT via the high SG pressure (MAX1) or high pressurizer pressure (MAX2) functions to prevent a pressure increase that would challenge the integrity of the RCPB. Also, EFW is actuated on a low SG level (MIN2) signal.

The U.S. EPR maintains the integrity of fission product barriers in the event of a PS failure by providing a diverse means to trip the reactor using either a high SG pressure, a high pressure pressurizer function, or a low SG narrow range level function. The U.S. EPR also provides a diverse means to trip the turbine following the diverse RT and a diverse means to initiate EFW on low SG level.

15.8.2.5 Loss of Offsite Power

The loss of nonemergency AC power results in the loss of power to the station auxiliaries. This situation could result from either a complete loss of the external grid (offsite) or a loss of the onsite AC distribution system. The loss of nonemergency AC power event causes the RCPs and MFW pumps to trip simultaneously at event initiation. This event causes a reactor coolant flow coastdown and a decrease in heat removal by the secondary system.

In the short term, the loss of forced circulation for the reactor coolant causes a sudden increase in the coolant temperature that could result in fuel damage. In the long term, the loss of MFW causes the pressure and temperature in the shell side of the SG to increase, which further increases the reactor coolant temperature and pressure. Both safety (EFW, MSRTs, and MSSVs) and non-safety (turbine bypass) systems are available to remove sensible and decay heat. The PS provides the primary RT via the low RCP speed or low RCS flow rate (two loops) functions to prevent a temperature increase that could cause fuel damage or a pressure increase that would challenge the integrity of the reactor coolant pressure boundary (RCPB). Also, EFW is actuated on a low SG level (MIN2) signal.

The U.S. EPR maintains the integrity of fission product barriers in the event of a PS failure by providing a diverse means to trip the reactor using a low RCS flow rate or a low SG narrow range level function. The U.S. EPR also provides a diverse means to trip the turbine following the diverse RT and a diverse means to initiate EFW on low SG level.

15.8.2.6 Closure of Main Steam Line Isolation Valves

The closure of one or more main steam line isolation valves (MSIV) interrupts the steam flow from the affected SGs to the turbine. Such events could be the result of a valve failure, a failure within the control system, or operator error. The closing stroke for an MSIV is slower than for a TSV, which tends to produce a less severe transient than the turbine trip. However, the location of the MSIV on the steam line is closer to the SG than the TSV. As a result, the volume available for pressurization is reduced, which acts to offset the effect of the slower stroke time.

The sudden reduction in steam flow causes the pressure and temperature in the secondary side of the SG to increase, which causes the reactor coolant temperature to increase, the coolant density to decrease, the water volume in the pressurizer to increase, and the reactor coolant pressure to increase. Several safety systems (e.g., EFW, MSRTs, and MSSVs) are available to remove sensible and decay heat. The PS provides the primary RT via the high SG pressure (MAX1) or high pressurizer pressure (MAX2) functions to prevent a pressure increase that would challenge the integrity of the RCPB. Also, EFW is actuated on a low SG level (MIN2) signal.

The U.S. EPR maintains the integrity of fission product barriers in the event of a PS failure by providing a diverse means to trip the reactor using a high SG pressure function or a high pressurizer pressure function. The U.S. EPR also provides a diverse means to trip the turbine following the diverse RT and a diverse means to initiate EFW on low SG level.

15.8.2.7 Rod Cluster Control Assembly Events

15.8.2.7.1 Uncontrolled RCCA Bank Withdrawal from Subcritical or Low Power Startup

An uncontrolled rod cluster control assembly (RCCA) bank withdrawal from subcritical or low power startup conditions causes an uncontrolled addition of reactivity to the reactor core that results in a power excursion. The neutron flux response to the continuous reactivity insertion is characterized by a very fast rise limited by the reactivity feedback effect of the negative fuel temperature coefficient. This self-limitation of the power excursion limits the power during the delay time for protection actions. The PS provides the primary RT via the high neutron flux (intermediate range) or low doubling time (intermediate range) functions to prevent a power increase that could cause fuel damage or a pressure increase that would challenge the integrity of the RCPB.

The U.S. EPR maintains the integrity of fission product barriers in the event of a PS failure by providing a diverse means to trip the reactor using a high neutron flux-power range function. The U.S. EPR also provides a diverse means to trip the turbine following the diverse RT and a diverse means to initiate EFW on low SG level.

15.8.2.7.2 Uncontrolled RCCA Bank Withdrawal at Power

An uncontrolled RCCA bank withdrawal at power causes an uncontrolled addition of reactivity to the reactor core that results in a power excursion. The neutron flux response to the continuous reactivity insertion is characterized by an increase in the core heat flux. The heat extraction by the SGs lags behind the core power generation until the SG pressure reaches the relief or safety valve setpoint; consequently, there is a net increase in the reactor coolant temperature and pressure. For slow reactivity insertion rates, the increase in the coolant temperature follows the nuclear power increase, which could result in fuel damage. The PS provides the primary RT via the high linear power density (HLPD), low departure from nucleate boiling ratio (DNBR), or excore high neutron flux rate functions to prevent a power increase that could cause fuel damage or a pressure increase that would challenge the RCPB integrity.

The U.S. EPR maintains the integrity of fission product barriers in the event of a PS failure by providing a diverse means to trip the reactor using a high neutron flux-power range function. The U.S. EPR also provides a diverse means to trip the turbine following the diverse RT and a diverse means to initiate EFW on low SG level.

15.8.2.7.3 Single RCCA Withdrawal at Power

The withdrawal of a single RCCA results in a continuous reactivity insertion which causes an increase in the average core power and temperature. The event also increases the local power peak in the zone where the RCCA has been withdrawn. The combination of penalizing thermal-hydraulic conditions and perturbed power distribution could result in fuel damage. The PS provides the primary RT via the HLPD or low DNBR functions to prevent a power increase that could cause fuel damage or a pressure increase that would challenge the RCPB integrity.

The U.S. EPR maintains the integrity of fission product barriers in the event of a PS failure by providing a diverse means to trip the reactor using a high neutron flux-power range function. The U.S. EPR also provides a diverse means to trip the turbine following the diverse RT and a diverse means to initiate EFW on low SG level.

15.8.3 Conclusion

Events subject to ATWS considerations were evaluated for potential damage to the barriers to fission product release. In each instance, the diverse trip functions and capabilities incorporated within the U.S. EPR design mitigate effects of an ATWS without compromising the integrity of the barriers to fission product release. Since the U.S. EPR design conforms to regulatory guidance by having a diverse scram system, the consequences of an AOO without an RT are bounded by the safety analyses in Chapter 15. Therefore, the ATWS success criteria listed in 10 CFR 50.46 and GDCs 12, 14, 16, 35, 38, and 50 are not required to be explicitly calculated.

The U.S. EPR design complies with 10 CFR 50.62, including the ATWS regulatory position for evolutionary reactor designs in SECY 90-016 (References 6 and 7) and SECY 93-087 (Reference 8), by minimizing the probability of an ATWS event through the use of diverse trip functions and capabilities. These capabilities include having equipment from sensor output to final actuation device diverse from the RT system to automatically initiate the EFW system and initiate a turbine trip under conditions indicative of an ATWS. Thus, the U.S. EPR design maintains the integrity of barriers to prevent a fission product release.

15.8.4 References

1. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article 3224, "Level C Service Limits," The American Society of Mechanical Engineers, 2004.

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4. NUREG/CR-5500, "Reliability Study: Combustion Engineering Reactor Protection System, 1984-1998," Vol. 10, U.S. Nuclear Regulatory Commission, November 2001.

5. NUREG/CR-5500, "Reliability Study: Babcock & Wilcox Reactor Protection System, 1984-1998," Vol. 11, U.S. Nuclear Regulatory Commission, November 2001.

6. SECY 90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," U.S. Nuclear Regulatory Commission, January 12, 1990.

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8. SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," U.S. Nuclear Regulatory Commission, April 2, 1993.

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