UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
OFFICE OF NEW REACTORS
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NRC REGULATORY ISSUE SUMMARY 2005-29, REVISION 1
ANTICIPATED TRANSIENTS THAT COULD DEVELOP INTO MORE SERIOUS EVENTS

ADDRESSEES

All holders of an operating license or construction permit for a pressurized water reactor (PWR) under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

All holders of and applicants for a PWR combined license, standard design approval, or manufacturing license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Reactors." All applicants for a PWR standard design certification, including such applicants after initial issuance of a design certification rule.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is revising Regulatory Issue Summary (RIS) 2005-29[1] to inform addressees of concerns identified during recent license amendment reviews. Specifically, licensing bases, as documented in final safety analysis reports (FSARs), updated FSARs (UFSARs), or design control documents (DCDs), failed to demonstrate that anticipated operational occurrences (AOOs, also Condition II events) would not progress to more serious events (Condition III or IV events). Revision 1 to RIS 2005-29 supersedes in its entirety RIS 2005-29. Neither the original RIS nor this revision transmits any new requirements or requires any specific action or written response.

BACKGROUND INFORMATION

Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," establishes minimum requirements for the principal design criteria for those water-cooled nuclear power plants that were licensed to these general design criteria (GDC). Those plants that were licensed before the GDCs may have similar plant-specific minimum requirements for the principal design criteria. The GDC that are applicable to the AOOs discussed in this RIS are the following:

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[1] Available in Agencywide Documents Access and Management Systems (ADAMS) under Accession No. ML051890212.

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- GDC 15, "Reactor Coolant System Design," requires that the reactor coolant system and its associated auxiliary control and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operations, including anticipated operational occurrences (AOOs).

- GDC 21, "Protection System Reliability and Testability," requires, in part, that the protection system be designed to assure that no single failure results in loss of the protection function.

- GDC 29, "Protection Against Anticipated Operational Occurrences," requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

Furthermore, 10 CFR 50.34(b), "Final safety analysis report," requires an FSAR to include among other things:

> A description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),"[2] provides guidance to applicants of or holders of all specific licenses or permits issued by the NRC. Chapter 15, "Accident Analyses," to RG 1.70 offers guidance as to which AOOs should be included in the safety analysis reports. The NRC issued similar guidance for combined licenses under 10 CFR Part 52 in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."[2]

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," referred to as the SRP[3], is intended to be a comprehensive and integrated document that provides the NRC staff with guidance that describes methods or approaches that the NRC staff has found acceptable for meeting NRC requirements. The NRC staff uses the categorization of accidents and transients as described in Chapter 15.0, "Introduction - Transient and Accident Analyses" of the SRP to review FSAR, or UFSAR, Chapter 15 safety analyses. Chapter 15.0, Section I.1.A of the SRP states, "Incidents of moderate frequency and infrequent events are also known as Condition II and Condition III events, respectively, in the commonly used, often cited but unofficial American Nuclear Society (ANS) standards." In other words, Condition II events (AOOs) are more frequent than Condition III events. Additionally, Condition II events should produce less severe consequences than Condition III events, as noted in Chapter 15.0 to the SRP:

[2] Publically available for download at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/rg/.
[3] Publically available for download at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/.

> If the risk of an event is defined as the product of the event's frequency of occurrence and its consequences, then the design of the plant should be such that all the AOOs and postulated accidents produce about the same level of risk (i.e., the risk is approximately constant across the spectrum of AOOs and postulated accidents). This is reflected in the general design criteria (GDC), which generally prohibit relatively frequent events (AOOs) from resulting in serious consequences, but allow the relatively rare events (postulated accidents) to produce more severe consequences.

Lastly, Chapter 15.0 to the SRP identifies acceptance criteria for the various categories of events. Of interest in this RIS is the following criterion applicable to Condition II events:

> By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV category without other incidents occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

For purposes of this RIS, this criterion will be referred to as the non-escalation criterion. The discussion in this RIS applies to those licensees who have incorporated the non-escalation criterion into the licensing bases (as documented in the plants' FSARs, UFSARs, or DCDs), either directly or by reference to an ANS standard[4]. The purpose of the non-escalation criterion is to prohibit the progression of the relatively high frequency Condition II events to more significant Condition III or Condition IV events. There have been several instances where the non-escalation criterion was not met, and a Condition II event generated a Condition III event. Specifically, inadvertent operation of the emergency core cooling system (ECCS) resulted in filling the pressurizer, and, in at least one case[5], relieved water through the power operated relief valves (PORVs). If water relief through the PORVs continues long enough, there is potential for the rupture disc on the pressurizer relief tank to rupture and RCS water to be spilled directly into containment, leaving the plant in a condition that will require significant clean-up efforts and potentially cause dose concerns.

During recent NRC review activities, the NRC staff discovered that several licensees had encountered difficulties in meeting the non-escalation criterion for several UFSAR Chapter 15 mass addition events[6]. Some of these difficulties were identified as the result of relying on

[4] Typically ANS 18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, August 6, 1973 or ANSI/ANS 51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," April 29, 1983.
[5] Available in ADAMS under Accession No. ML051860338.
[6] Mass addition events can increase reactor coolant system (RCS) inventory. Mass addition events do not contribute to fuel clad damage since they do not produce conditions (e.g., RCS depressurization at a high power level) that would tend to reduce core thermal margin (i.e., departure from nucleate boiling ratio (DNBR)).

guidance provided by Westinghouse, contained in Nuclear Safety Advisory Letter, NSAL 93-013[7], and its supplement. NSAL 93-013, a third-party guidance document, has not been approved for use[8] nor endorsed by the NRC.

SUMMARY OF ISSUE

The NRC staff observed that certain safety analyses, specifically in Chapter 15 of UFSARs, failed to meet the non-escalation criterion for three mass addition Condition II events (AOOs). These Condition II events are: the chemical and volume control system (CVCS) malfunction, the inadvertent operation of the ECCS during normal operation, and the inadvertent opening of a PORV or a pressurizer safety valve (PSV). The NRC staff is concerned that these Condition II events (AOOs) could escalate to a small break loss of coolant accident (SBLOCA), a Condition III event, in turn violating the non-escalation criterion. Several of these licensing basis safety analyses that failed to meet the non-escalation criterion were identified as having relied upon information from the third-party guidance provided in NSAL 93-013 and its supplement. One of the failures resulted in NRC follow-up actions to correct the issue. The below sections describe the NRC staff concerns in regard to meeting the non-escalation criterion for the following three licensing basis analyses: CVCS malfunction that increases reactor coolant inventory, inadvertent operating of the ECCS (IOECCS), and inadvertent opening of a PORV or PSV.

A. CVCS Malfunction That Increases Reactor Coolant Inventory

The licensing basis analysis postulates the CVCS to malfunction due to a component fault (e.g., a water level sensor failing low) or an operator error. The result is high-rate charging flow injection into the RCS, which could result in the pressurizer going water-solid and the PORVs passing water. Without operator action to terminate the event, the CVCS malfunction, a Condition II event, could become a more serious Condition III SBLOCA event. The NRC staff has observed two different licensing basis analyses for the CVCS malfunction event that failed to meet the non-escalation criterion.

A.1 <u>Evaluation of the CVCS malfunction is not required, since it is already analyzed in the licensing basis (i.e., as a reactivity anomaly)</u>

The CVCS malfunction is listed in RG 1.70 as a mass addition event and also as a reactivity anomaly. However, these cases are not equivalent and not interchangeable.

The licensing basis analysis of a CVCS malfunction as a reactivity anomaly (i.e., boron dilution) event relies on the protection of various neutron flux-based trips and operator actions to end the that the operators have more than 15 minutes to end the boron dilution prior to returning to criticality.

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[7] The supplement to NSAL 93-013 can be found as Enclosure 2 under Accession No. ML050320117.
[8] With respect to third-party documents, “approved for use” means that the NRC has determined that the matters addressed in the document are technically acceptable and consistent with NRC regulatory requirements, guidance and policy, but the NRC neither supports nor discourages entities from using the positions set forth in the document. “Endorse,” by contrast means that in addition to the NRC determination above, the NRC encourages entities to follow the positions in the document.

However, the licensing basis analysis of a CVCS malfunction as a mass addition event does not assume minimum RCS volume, a neutron flux-based trip, or focus on reactivity concerns. Rather, the focus is on the integrity of the RCS. The charging pumps add water to the RCS, and power generation continues until a reactor trip signal is produced by the automatic reactor protection system (e.g., a high pressurizer water level trip signal). However, the charging flow continues to fill the pressurizer and can only be stopped by operator action.

Because the mass addition licensing basis analysis and the reactivity anomaly licensing basis analysis of the Condition II CVCS malfunction event have different initial conditions and goals, only performing the reactivity anomaly licensing basis analysis does not meet the non-escalation criterion.

A.2 <u>Evaluation of the CVCS malfunction is not required, since it is not as severe as the IOECCS</u>

A reactor trip signal is generated during the CVCS malfunction, and the event is terminated when the operator shuts off charging flow. The CVCS malfunction could be less limiting (i.e., the pressurizer is predicted to fill at a slower rate) than the IOECCS. The time to fill the pressurizer could be longer, since the coolant shrinks when the reactor is tripped, and the charging flow rate is lower than when the charging pumps are operating as part of the ECCS. However, the IOECCS is a mass addition event that begins with a reactor trip, whereas the CVCS malfunction involves the addition of both mass and heat to the RCS, making a direct comparison between the two events impossible. Therefore, a statement in the licensing basis (usually FSAR or UFSAR) that the CVCS malfunction is not as severe as the inadvertent ECCS evaluation does not meet the non-escalation criterion. The licensing basis analysis of the Condition II CVCS malfunction event is necessary in order to demonstrate that the non-escalation criterion has been met.

B. IOECCS

The licensing basis postulates IOECCS to occur as the result of a spurious safety injection signal. ECCS is assumed to operate the ECCS pumps at their peak performance level (i.e., no failures are assumed). The shutoff head of most ECCS is greater than the nominal RCS pressure and could lift the PORVS or PSVs. If one or more PORVs open while the pressurizer is water-solid, then the PORV(s) are generally assumed to fail open, since valves that are not qualified to relieve water are conservatively assumed to remain in the fully open position. In this case, a stuck-open PORV could cause a Condition III SBLOCA event and fail to meet the non-escalation criterion. Five of the several alternative approaches suggested by NSAL 93-013, in regards to the IOECCS, failed to meet the non-escalation criterion and are discussed in detail below.

B.1 <u>Closing a block valve to isolate a stuck-open PORV</u>

Licensing basis analyses that predict that the pressurizer fills and water is relieved through the PORVs often cite the following from NSAL 93-013 in their UFSARs, “Water relief through the PORVs is not a concern, because the PORV block valves can be used to isolate the PORVs if they fail to close." The NRC staff noted in the original version of RIS 2005-29 that closing a block valve to isolate a stuck-open PORV is an action that would be taken to respond to a Condition III loss of coolant accident, not to a Condition II IOECCS. This demonstrates that the Condition II IOECCS does in fact become a Condition III event because a stuck open PORV is expected as the result of the event and not as an independent failure. Therefore, this approach does not demonstrate that the Condition II IOECCS meets the non-escalation criterion.

B.2 Application of the PSVs as a protection system

Licensing basis analyses that are based upon the application of water-qualified PSVs to mitigate IOECCS require the PSVs to open, relieve water, and reseat. The rationale relies upon the premise that none of the PORVs will open, or if they open and stay open, then they can be closed or isolated citing the same flawed logic as approach B.1 above. The rationale for this licensing basis analysis may also assume that the PORVs are not operable, which the NRC staff has identified as a non-conservative assumption[9]. It is conservative to maximize the rate at which the pressurizer fills during an IOECCS. This is done by assuming that the pressurizer PORVs and sprays are operable since they tend to limit the rate of RCS pressurization, which would permit a relatively higher rate of ECCS delivery. Thus, the pressurizer fills more rapidly as steam is relieved through the PORVs.

In general, the NRC staff considers the substitution of PSVs for PORVs to be inappropriate for Condition II events[10]. The PORVs, which are part of the automatic pressure control system, along with the pressurizer heaters and spray, are used to maintain the plant within its acceptable operating range. They limit RCS pressurization to levels below the high pressure reactor trip setpoint and prevent unnecessary operation of the PSVs. Condition II events are normally handled by the automatic pressure control system (e.g., during load rejections). If the plant conditions exceed the capabilities of the automatic pressure control system during a Condition II event, then the event should be ended with, at most, a reactor shutdown, as specified in the design requirements for Condition II events (AOOs)[11]. In contrast, PSVs will not open until after the reactor has been tripped, since the opening setpressure for the PSVs is higher than the high pressure reactor trip setpressure. In other words, PSVs are not expected to open during Condition II events. Furthermore, mass addition events cannot pressurize the RCS to challenge the RCS pressure safety limit. The delivery of ECCS water effectively ends when the RCS pressure reaches the ECCS pump shutoff head (about 2600 psia, which is well

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[9] Chapter 15.5.1-15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory."

[10] Note 8 of Matrix 8 of Section 2.1 to Review Standard (RS) 001, "Review Standard for Extended Power Uprates," states:
For the inadvertent operation of emergency core cooling system and chemical and volume control system malfunctions that increase reactor coolant inventory events: (a) non-safety grade pressure-operated relief valves should not be credited for event mitigation and (b) pressurizer level should not be allowed to reach a pressurizer water-solid condition.
RS-001 is publically available for download at http://www.nrc.gov/reactors/operating/licensing/power-uprates/related-reg-guides/guides-pu.html.

[11] Most FSARs, or UFSARs, state, directly or by reference to ANS Standards, "Condition II events shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action."

below the RCS pressure safety limit of 110 percent of RCS design pressure). Therefore, as stated before in RIS 2005-29, this approach does not adequately demonstrate that IOECCS will meet the non-escalation criterion.

B.3 <u>A stuck-open PORV or PSV resulting from an IOECCS is already addressed as an inadvertent opening of a PORV or PSV</u>

Some licensing basis analyses claim that an IOECCS that results in a stuck-open PORV need not be considered as an escalation to the Condition III category because an inadvertent opening of a PORV or PSV is already analyzed as a Condition II event in most licensing bases.

However, the inadvertent opening of a PORV or PSV is often analyzed in the licensing basis to show that adequate, timely protection is provided by an automatic reactor trip, a different Condition II design requirement[12]. The licensing basis analysis typically predicts the reactor trip to occur in a few seconds, which is before the ECCS starts injecting into the RCS. Consequently, the Condition II inadvertent opening of a PORV or PSV licensing basis analysis does not model ECCS injection, which is an initial condition for IOECCS. Therefore, inadvertent opening of a PORV or PSV cannot be used to bound IOECCS and does not meet the non-escalation criterion for a stuck open PORV or PSV resulting from an IOECCS.

B.4 <u>A stuck-open PORV or PSV is not as severe as a SBLOCA</u>

Licensing basis analyses that consider a stuck-open PORV or PSV to be less severe than a SBLOCA do not fit into the classic scheme of event classification, upon which nuclear safety analyses are based, and effectively create a new category of high-frequency, severe-consequence events. Comparing or bounding the licensing basis IOECCS to or with any of the SBLOCA licensing basis analyses does not meet the non-escalation criterion since the IOECCS is a Condition II event and a SBLOCA is a Condition III event.

B.5 <u>RCS inventory that exits through the PORV(s) is made up by ECCS flow</u>

Some licensees have included the following statements from NSAL 93-013 about Condition II events into the licensing basis:

1. "... a Condition II event as a minor reactor coolant system leak which would not prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems only," and

2. "Since the cause of the water relief is the ECCS flow, the magnitude of the leak will be less than or equivalent to that of the ECCS (i.e., operation of the ECCS maintains RCS inventory during the postulated event and establishes the magnitude of the subject leak)."

[12] Regarding the acceptance criteria necessary to meet the requirements of GDCs 10, 15, 20, and 26 for incidents of moderate frequency, the SRP states in Chapter 15.0, "Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs…"

The NRC staff identified two issues with how these two statements were applied in IOECCS licensing basis analyses. First, the charging system, when operated as a part of the ECCS, cannot be considered to be a normal makeup system. Charging flow operated as a part of the ECCS is not controlled by a pressurizer level program or by letdown flow rates. It operates at maximum capacity, and it does not shut down until an operator shuts it down. When the charging system is operated as a part of the ECCS, its purpose is to supply emergency core cooling, not to maintain a programmed pressurizer level.

Second, the water relief rate through the PORVs will be critical flow, which is determined by the pressure difference between the pressurizer and the pressurizer relief tank (e.g., about 2300 psi), and ultimately, the containment. In the short term, the water flowing out of the RCS, through the failed PORV(s) or PSV(s), far exceeds the rate of water flowing into the RCS from the ECCS, depleting the RCS inventory. As the open PORVs depressurize the RCS, more ECCS flow will be delivered by the ECCS pumps. Eventually, the RCS pressure drops down to a few hundred pounds and the flow leaving the RCS through the PORVs will be equal to the flow entering the RCS from the ECCS. Application of these two statements to the Condition II IOECCS licensing basis analysis does not meet the non-escalation criterion.

C. Inadvertent Opening of a PORV or PSV

The primary objective of the licensing basis analysis regarding inadvertent opening of a PORV or PSV is to show that the event would not lead to fuel clad damage. The licensing basis analysis typically demonstrates that the automatic reactor trip is demanded by the portion of the reactor protection system that is designed to protect against fuel clad damage (e.g., overtemperature ΔT trip or low pressure trip). Since the reactor trip ends the degradation of thermal margin, licensing basis analyses of this event tend to not extend past the time of reactor trip to show that there would be no fuel clad damage. Although the reactor shutdown protects against fuel clad damage, it does not end the RCS depressurization. Manual action must be taken to close the inadvertently opened PSV, PORV or its block valve before actuation of ECCS could begin. An actuation of the ECCS could lead to a water-solid pressurizer, followed by water relief through the PORVs, and ultimately to a Condition III SBLOCA event. Most licensing basis analyses choose to show closure of the PSV, PORV, or its block valve prior to ECCS actuation, in order to avoid the pressurizer filling shortly after ECCS delivery begins. One approach within NSAL 93-013 failed to meet the non-escalation criterion and is described below.

C.1 <u>Closure of the PSV, PORV, or block valve after ECCS actuation</u>

Failure to close the opened PSV, PORV, or its block valve before the ECCS actuation setpoint is reached could result in a scenario where the ECCS introduces a large quantity of water into a depressurizing RCS, causing the pressurizer to fill rapidly. The spuriously opened PSV or PORV could relieve water and stick open if it is not water qualified. Once the ECCS is in operation, it could be expected to remain in operation for several minutes since the operators need time to follow the procedures governing the shut off of ECCS. After the ECCS flow is terminated, operators could begin closing the PSV, PORV, or its block valve. If the PSV, PORV, or its block valve do not close, then the inadvertent opening of a PSV or PORV could continue as a Condition III SBLOCA and fail to meet the non-escalation criterion for Condition II events.

BACKFITTING AND ISSUE FINALITY

This RIS revision does not present a new or changed NRC staff position or guidance with respect to progression of anticipated operational occurrences, nor does it require any action or written response from any addressee. Therefore, issuance of this revised RIS is not a backfit under 10 CFR 50.109, "Backfitting", nor does it constitute a violation of any issue finality provision in 10 CFR Part 52.

***FEDERAL REGISTER* NOTIFICATION**

[Discussion to be provided in final RIS]

CONGRESSIONAL REVIEW ACT

[Discussion to be provided in final RIS]

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not contain new or amended information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing requirements were approved by the Office of Management and Budget (OMB), approval number 3150-0011 and 3150-0151.

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