U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

15.5.1 INADVERTENT OPERATION OF ECCS AND REACTOR COOLANT INVENTORY 15.5.2 AND PURIFICATION SYSTEM (RCI) MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of transient and accident analyses for pressurized-water reactors

Secondary - None

I. AREAS OF REVIEW

In the mPower™ integrated pressurized-water reactors (PWR) design, there are eight reactor coolant pumps (RCPs) located around the Pressurizer. Inside the reactor pressure vessel (RPV), there are two flow dividers. The flow divider directs the reactor coolant system (RCS) flow to the RCPs and the RCPs discharge the flow to the bottom of the RPV through the downcomers essentially forming circulation flow in the RPV. These pumps are expected to be powered from off-site power sources and may be powered from more than a single source. Out of eight total pumps, some may be tripped due to loss of power supply or due to mechanical problems. There is a potential for startup of an inactive pump or pumps at an incorrect temperature and flow controller malfunction causing an increase in core flow.

Certain anticipated operational occurrences (AOOs) can cause an unplanned increase in RCS inventory. Spurious passive core cooling injection from the ECC system is possible. The RCI system provides high-pressure reactor coolant makeup. The RCI pumps may be started spuriously and inject water into the reactor. Depending on the temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. The reactor will trip from high water level, high flux, high pressure, low pressure, or from a safety injection signal.

If the AOO that causes an unplanned increase in RCS inventory is a spurious actuation of the emergency core cooling system (ECCS), then the reactor may be automatically tripped by the ECCS actuation signal. The ECCS, once started, is not stopped by receipt of an automatic signal. Manual action, taken according to Emergency Operating Procedures, is required to stop the ECCS.

It is important to consider the inadvertent operation of ECCS and inadvertent start-up of RCI pumps.

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The review of events leading to an increase in reactor coolant inventory considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and especially operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff, and have been applied in accordance with any limitations that may have been specified in the staff's acceptance. If a referenced analytical method has not been previously reviewed, the reviewer initiates a generic evaluation of the new analytical model. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of those transients analyzed are reviewed to ensure that the consequences meet the acceptance criteria given in Subsection II, below.

Further, the results of the analysis are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

For a design certification (DC) application, the review will also address combined license (COL) action items and requirements and restrictions (e.g., interface requirements and site parameters).

For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g. interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other DSRS sections interface with this section as follows:

1. General information on transient and accident analyses is provided in DSRS Section 15.0.

2. Design basis radiological consequence analyses associated with design basis accidents are reviewed under DSRS Section 15.0.3.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. General Design Criterion (GDC) 10, which require that the reactor core and associated coolant control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

2. GDC 13, which requires, in part that the effect of instrumentation shall be provided to monitor variables and systems over their anticipated ranges for anticipated operational occurrences to assure adequate safety. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, which 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.

4. GDC 26, which requires, in part, the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded under conditions of normal operation, including anticipated operational occurrences, with appropriate margin for malfunctions, such as stuck rods.

5. For plants with licensing bases that incorporate RG 1.70, ANS 51.1 (for PWRs), or ANSI/ANS-52.1-1978 (for boiling water reactors), there are acceptance criteria, in DSRS 15.0, “Condition II” events, or events of moderate frequency, “Condition III events, or infrequent events, and “Condition IV” events, or postulated accidents of low probability. Acceptance criteria are also defined for Condition II, III, and IV events. Regulatory Issue Summary (RIS) 2005-29, which relates to the escalation of a Condition II event into a Condition III or IV event, is also applicable to these plants.

The basic objectives in reviewing the events leading to an increase in reactor coolant inventory are:

1. To identify which of the AOOs leading to an RCS inventory increase are the most limiting.

2. To verify that, for the most limiting transients, the plant responds to the RCS inventory increase in such a way that the criteria regarding fuel damage, RCS pressure, and escalation to a more serious event are met.

<u>DSRS Acceptance Criteria</u>

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the U.S. Nuclear Regulatory Commission’s (NRC’s) regulations identified above are set forth below. The DSRS is not a substitute for the NRC’s regulations, and compliance with it is not required. Identifying the differences between this DSRS section and the design features, analytical techniques, and procedural measures proposed for the facility, and discussing how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria, is sufficient to meet the intent of 10 CFR 52.47(a)(9), “Contents of applications; technical information.” The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41) for COL applications.

1. This event is an AOO, as defined in 10 CFR 50, Appendix A. Acceptance criteria for AOOs are specified in DSRS 15.0.

The specific acceptance criteria derived from GDC 10, 13, 15, and 26, and from the aforementioned American Nuclear Society (ANS) standards are:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

2. Fuel cladding integrity should be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

3. An AOO should not generate a more serious plant condition without other faults occurring independently.

The applicant's analysis of events leading to an increase of reactor coolant inventory should be performed using an acceptable analytical model. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer performs an evaluation of the new method as part of its review under this DSRS section.

The values of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model:

1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

2. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core.

3. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

<u>Technical Rationale</u>

The technical rationale for application of these acceptance criteria to the areas of review addressed by this DSRS section is discussed in the following paragraphs:

1. Compliance with GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to ensure that fuel design limits are not exceeded during any condition of normal operation, including AOOs.

   The requirements of GDC 10 apply to the RCI malfunction events, since power level could conceivably increase as water is being added to the RCS, until a reactor trip condition is reached.

   Meeting this criterion provides reasonable assurance that AOOs will not result in fuel damage and subsequent fission product release.

2. Compliance with GDC 13 requires the provision of instrumentation that is capable of monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions as appropriate, to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequence of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

3. Compliance with GDC 15 requires that the reactor coolant system and 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 operation, including AOOs.

   The requirements of GDC 15 apply to mass addition events since the additional RCS inventory could pressurize the RCS. Without a significant addition of heat to the RCS water, the RCS would not pressurize to levels exceeding the shutoff head of the ECCS pumps, and therefore would not be expected to violate the RCS pressure safety limits. Similarly, without power generation (i.e., after reactor trip) the main steam system would not be likely to pressurize beyond the steam line safety valve setpoint levels.

   Meeting this criterion provides reasonable assurance that AOOs will not result in damage to the reactor coolant pressure boundary and subsequent fission product release.

4. Compliance with GDC 26 requires reactivity control system redundancy and capability. The requirements of GDC 26 apply to this section since the appropriate mitigation for an AOO is a reactor trip (and in this case, manual action to end the mass addition to the RCS). Once shut down, the reactor should remain in a shutdown condition.

   Meeting this criterion provides reasonable assurance that anticipated operational occurrences will not result in fuel damage and subsequent fission product release.

5. By meeting the ANS design requirement that states, "by itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV type without other incidents occurring independently," or by satisfying the corresponding criterion of DSRS Section 15.0 for AOOs (i.e., an AOO cannot generate a postulated accident without other incidents occurring independently). Such compliance limits the probability of initiating any of the more safety significant events at a relatively high frequency (i.e., one or more incidents prevent the event from developing into a more serious event occurring during the lifetime of the plant).

   The mass addition events, the inadvertent operation of ECCS (DSRS Section 15.5.1), and the RCI malfunction that increases reactor coolant inventory and (DSRS Section 15.5.2), are more likely to challenge this criterion than other AOOs. The inadvertent operation of ECCS is a concern only in plants that are equipped with charging pumps that are used in the ECCS mode that can deliver flow to the RCS when the RCS is at nominal pressure.

The inadvertent operation of an ECCS is postulated to occur as the result of spurious safety injection (SI) signal. The SI signal trips the reactor and actuates the ECCS. Therefore, since the reactor is shut down throughout the transient, violation of the DNBR safety limit is not likely to be a concern. The inadvertent operation of an ECCS, that actuates charging pumps in an SI mode, is assumed to operate the charging system at its peak performance level (i.e., no failures are assumed). The shutoff head of the charging system is necessarily greater than the nominal RCS pressure, and possibly high enough to lift the pressurizer safety valves; but not high enough to pressurize the RCS to 110 percent of its design pressure (e.g., 2750 psia). Therefore, overpressurization of the RCS is not likely to be a concern.

Because the inadvertent operation of ECCS causes an immediate reactor trip, there is no power/steam flow mismatch. Consequently, there is little or no effect upon the main steam system. Overpressurization of the main steam system, therefore, is not likely to be a concern.

If the inadvertent operation of the ECCS causes one or more pressurizer power-operated relief valves (PORVs) to open while the pressurizer is water-solid, then the PORV is generally assumed to fail open (i.e., PORVs are assumed to fail in the open position after having relieved water, if they are not (1) safety-related equipment and (2) qualified for water relief). The inadvertent operation of the ECCS, therefore, can lead to a loss-of-coolant accident, which may be considered an AOO, a postulated accident, a Condition II event, or a Condition IV event, depending upon break size and the event categorization scheme in the licensing basis (see DSRS Section 15.0).

Typically, design basis accident analyses show that AOO cannot become a more serious event, by demonstrating that the pressurizer does not become water-solid at any time during the transient, and therefore, a PORV cannot ever relieve water. The event ends when the charging flow is terminated by the operator. The analysis objective is to show that the pressurizer does not become water-solid before the operator can terminate the transient,usually at about ten minutes (or longer) after the event begins. If the plant is equipped with PORVs that are (1) safety-related equipment and (2) qualified for water relief, then they may be assumed to reseat properly after having relieved water. The pressurizer safety valves, too, may be assumed to reseat properly after having relieved water; but only if such valves have been qualified for water relief.

It is conservative to assume that PORVs open and relieve steam in order to limit the RCS pressurization, and thereby increase the charging flow rate (and the resulting pressurizer fill rate). This shortens the time available to the operator to terminate the charging flow before the pressurizer fills.

Unlike the inadvertent operation of the ECCS, the RCI malfunction that increases reactor coolant inventory (see DSRS Section 15.5.2), a related AOO, does not lead directly to a reactor trip. The reactor may be tripped automatically, from a signal that is generated during the transient, e.g., high pressurizer pressure or level. Since power is being generated prior to reactor trip, the event could cause departure from nucleate boiling to occur. However, this is not likely, since (1) core pressure increases, and (2) the reactor protection system automatically trips the reactor when it senses a reduction in thermal margin. Like the inadvertent operation of ECCS, the RCI malfunction is not expected to pose a concern with respect to RCS and main steam system overpressurization.

The RCI malfunction event should meet the same acceptance criteria as the inadvertent operation of ECCS. The event is mitigated (i.e., terminated) when the operator shuts off the charging flow. The RCI malfunction event is expected to be less limiting (i.e., to fill the pressurizer more slowly) than the inadvertent operation of ECCS event. This is due to some coolant shrinkage that occurs when the reactor is tripped, and to the lower charging flow rate that is delivered when the charging pumps are not operating as part of the ECCS.

III. REVIEW PROCEDURE

These review procedures are based on the identified DSRS acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Programmatic Requirements - In accordance with the guidance in NUREG-0800 "Introduction," Part 2 as applied to this DSRS Section, the staff will review the programs proposed by the applicant to satisfy the following programmatic requirements. If any of the proposed programs satisfies the acceptance criteria described in Subsection II, it can be used to augment or replace some of the review procedures. It should be noted that the wording of "to augment or replace" applies to nonsafety-related risk-significant structures, systems, and components (SSCs), but "to replace" applies to nonsafety-related nonrisk-significant SSCs according to the "graded approach" discussion in NUREG-0800 "Introduction," Part 2. Commission regulations and policy mandate programs applicable to SSCs that include:

    A. Maintenance Rule SRP Section 17.6 (DSRS Section 13.4, Table 13.4, Item 17, Regulatory Guides 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." and RG 1.182; "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants".

    B. Quality Assurance Program SRP Sections 17.3 and 17.5 (DSRS Section 13.4, Table 13.4, Item 16).

    C. Technical Specifications (DSRS Section 16.0 and SRP Section 16.1) – including brackets value for DC and COL. Brackets are used to identify information or characteristics that are plant specific or are based on preliminary design information.

    D. Reliability Assurance Program (SRP Section 17.4).

    E. Initial Plant Test Program (RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants, "DSRS Section 14.2, and DSRS Section 13.4, Table 13.4, Item 19).

    F. Inspections, tests, analyses, and acceptance criteria (DSRS Chapter 14).

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), for new reactor license

applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety and medium- and high-priority generic safety issues that are identified in the version of NUREG-0933 current on the date six months before application and that are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Reference: 10 CFR 52.47(a)(21), 10 CFR 52.47(a)(22) , and 10 CFR 52.47(a)(8), respectively. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report section.

The applicant's description of events leading to an increase in reactor coolant inventory is reviewed with respect to the occurrences leading to the initiating event. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to determine the following:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. Credit taken for the functioning of normally operating plant systems;

4. Operation of required engineered safety systems;

5. The extent to which operator actions are required; (Note: an operator action to shut off the charging pump flow is normally required to terminate these AOOs.

6. That appropriate margin for malfunctions, such as stuck rods (see subsection II.3.b), is taken into consideration.

The applicant should present a quantitative analysis in the SAR of the most limiting events that lead to an increase in reactor coolant inventory. Such an analysis should demonstrate that AOOs could not develop into more serious events. The reviewer examines the timing of the initiation of those protection and engineered safety systems, and operator actions needed to limit the consequences of the event to acceptable levels. The reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed to determine if these models have been previously reviewed and found acceptable by the staff. If not, a generic review of the models is initiated.

The values of system parameters and initial core and system conditions used as input to the model are reviewed. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that the selected core burnup yields the minimum margins.

The results of the applicant's analysis are reviewed in accordance with the acceptance criteria presented in subsection II regarding maximum pressure in the reactor coolant and main steam systems, the minimum critical heat flux ratio (MCHFR) DNBR and the possibility of escalation to a more serious event. The variations with time during the transient of the neutron power, heat fluxes (average and maximum), reactor coolant system pressure, minimum DNBR (PWR)); core and recirculation coolant flow rates, coolant conditions (inlet temperature, core average temperature (PWR), average exit and hot channel exit temperatures, and steam fractions), steam line pressure, containment pressure, pressure relief valve flow rate, and flow rate from the reactor coolant system to the containment system are reviewed, as applicable. The review will also compare values of the more important of these parameters for the events leading to an increase in reactor coolant inventory with those predicted for other similar plants to confirm that they are within the expected range.

For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit or other NRC approvals (e.g. manufacturing license, site suitability report or topical report).

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff's technical review and analysis, as augmented by the application of programmatic requirements in accordance with the staff's technical review approach in the DSRS Introduction, support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

The staff concludes that the analysis of a transient resulting in an increase in reactor coolant inventory is acceptable and meets the requirements of General Design Criteria 10, 15, and 26, and the guidance of ANS standards. This conclusion is based on the following:

1. In meeting GDC 10, 13, 15, and 26 as discussed below, the staff has determined that the applicant's analysis was performed using a mathematical model that has been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The staff has further determined that the positions of RG 1.53 for the single-failure criterion and RG 1.105 for instruments have also been satisfied.

2. The applicant has met the requirements of GDC 10, and 26 with respect to demonstrating that resultant fuel damage is maintained because the specified acceptable fuel design limits were not exceeded for this event.

3. The applicant has met the GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and

manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

4. The applicant has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded by this event and that resultant leakage will be within acceptable limits. This requirement has been met since the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

5. The applicant has met the requirements of GDC 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margins for malfunctions because the specified acceptable fuel design limits were not exceeded.

6. The applicant has satisfied the ANS design criteria that prohibits the escalation of an AOO to a more serious incident without other incidents, occurring independently.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this DSRS section.

V. IMPLEMENTATION

The staff will use this DSRS section in performing safety evaluations of mPower™-specific DC, or COL, applications submitted by applicants pursuant to 10 CFR Part 52. The staff will use the method described herein to evaluate conformance with Commission regulations.

Because of the numerous design differences between the mPower™ and large light-water nuclear reactor power plants, and in accordance with the direction given by the Commission in SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (ML102510405), to develop risk-informed licensing review plans for each of the small modular reactor reviews including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower™ -specific DC, or COL submitted pursuant to 10 CFR Part 52 to comply with 10 CFR 52.47(a)(9), "Contents of applications; technical information."

This regulation states, in part, that the application must contain "an evaluation of the standard plant design against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application." The content of this DSRS section has been accepted as an alternative method for complying with 10 CFR 52.47(a)(9) as long as the mPower™ DCD FSAR does not deviate significantly from the design assumptions made by the NRC staff while preparing this DSRS section. The application must identify and describe all differences between the standard plant design and this DSRS section, and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria. If the design assumptions in the DC application deviate significantly from the DSRS, the staff will use the SRP as specified in 10 CFR 52.47(a)(9). Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design assumptions. The same approach may be used to meet the requirements of 10 CFR52.79(a)(41), and COL applications.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criteria.

2. RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

3. ANS 51.1, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor [PWR] Plants,” 1983 (replaces ANSI N18.2-1973) (withdrawn in 1998).

4. ANSI/ANS-52.1-1983, “Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants,” (withdrawn in 1998).

5. RIS 2005-29, “Anticipated Transients that could Develop into More Serious Events,” December 14, 2005 (ADAMS Accession No. ML051890212).

6. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

7. RS-001, Revision 0, "Review Standard for Extended Power Uprates," December 2003, (Note 8 of Matrix 8 of Section 2.1).