NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

15.0 INTRODUCTION - TRANSIENT AND ACCIDENT ANALYSES

REVIEW RESPONSIBILITIES

Primary - Organizations responsible for review of transient and accident analyses ~~for PWRs/BWRs~~, and design-basis accident radiological consequence analyses for light-water power reactors

Secondary ~~-~~– None

The evaluation of the safety of a nuclear power plant requires analyses of the plant's responses to postulated equipment failures or malfunctions. Such analyses help to determine the limiting conditions for operation, limiting safety system settings, and design specifications for ~~safety-related~~ components and systems to protect public health and safety. These analyses are a focal point of ~~the license amendment request (LAR),~~reviews of standard design ~~certification (DC), and~~ ~~combined license (COL) reviews~~license applications under Title 10 of the *Code of Federal Regulations* (10 CFR) Parts 50 and 52.

I. AREAS OF REVIEW

The specific areas of review are as follows:

1. Categorization of Transients and Accidents. The reviewer ensures that the applicant's selection and assembly of the plant transient and accident analyses represent a sufficiently broad spectrum of transients and accidents~~,~~ or initiating events.

   Initiating events are categorized as transients (i.e., anticipated operational occurrences (AOOs)) or accidents according to their expected frequency of occurrence and by type. Categorization by frequency of occurrence provides a basis for selection of the applicable analysis acceptance criteria for each initiating event. ~~Categorization of~~

This Standard Review Plan~~,~~ (SRP), NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission (NRC) staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The ~~Standard Review Plan~~SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

The ~~standard review plan~~SRP sections are numbered in accordance with corresponding sections in Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of ~~Regulatory Guide~~RG 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on ~~Regulatory Guide~~RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.

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~~initiating~~

Initiating events are further categorized by the type of impact they have upon the reactor system. Categorization by type provides a basis for comparison between events with similar plant responses, which ~~makes it possible to identify~~facilitates identification and ~~evaluate~~evaluation of the limiting cases (i.e., the cases that can challenge the analysis acceptance criteria).

A. Categorization According to Frequency of Occurrence. Each initiating event is categorized as either an ~~anticipated operational occurrence~~ (AOO) ~~or as~~, a postulated accident, or a beyond-design-basis event.

AOOs, as defined in Appendix A to 10 CFR Part 50, are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.

~~The SRP uses the term AOOs to refer to the events that are categorized in Regulatory Guide 1.70 and in Regulatory Guide 1.206 as~~Note the regulations and the SRP use the term AOO while RG 1.70 (Ref. 2) uses the terms incidents of moderate frequency (i.e., events that are expected to occur several times during the plant's lifetime) and infrequent ~~events~~incidents (i.e., events that may occur during the lifetime of the plant).

~~Incidents of moderate frequency and infrequent events are also known as Condition II and Condition III events, respectively, in~~ For facilities that were licensed using different categorizations, the ~~commonly used, oft- cited but unofficial American Nuclear Society (ANS) standards. The~~ reviewer will continue to evaluate applications~~,~~ according to ~~the categorizations and acceptance criteria of References 4 and 5, for licensees that have these categorizations in~~ their licensing bases, ~~or if they wish, according~~unless the licensee proposes to adopt the categorizations and acceptance criteria of this SRP section. The reviewer will evaluate new applications (i.e., those pertaining to plants that are not yet constructed) according to the categorizations and acceptance criteria of this SRP section.

The reviewer ensures the applicant or licensee has conducted a systematic and comprehensive search for initiating events based on plant-specific design features and site characteristics. The events specified in the relevant 10 CFR Part 50, Appendix A, General Design Criteria (GDC) identified in this SRP establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission when the GDC were promulgated. For some water-cooled nuclear power plants, the GDC may not be sufficient and additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the GDC may not be necessary or appropriate. For plants such as these, departures from the GDC must be identified and justified.

The following are ~~some~~ examples of AOOs ~~in~~ considered in previous licensing reviews for operating pressurized-water reactor (PWR) and boiling-water reactor (BWR) designs:

- Inadvertent control rod or rod group withdrawal (PWR and BWR)
- Loss or interruption of core coolant flow, excluding reactor coolant pump locked rotor (PWR)
- Inadvertent moderator cooldown (PWR and BWR)
- Inadvertent chemical shim dilution (PWR)
- Depressurization by spurious operation of an active element, such as a relief valve (PWR and BWR)
- Reactor-turbine load mismatch, including loss of load and turbine trip (PWR and BWR)
- Blowdown of reactor coolant through a safety valve (PWR and BWR)
- Control rod drop (inadvertent addition of absorber) (PWR)
- Loss of normal feedwater (PWR and BWR)
- Loss of condenser cooling (PWR and BWR)
- ~~Steam generator tube leaks (PWR)~~

- ~~Reactor-turbine load mismatch, including loss of load and turbine trip (PWR and BWR)~~

- ~~Control rod drop (inadvertent addition of absorber) (PWR)~~

- Single error of an operator (PWR and BWR)

- ~~Single failure~~Failure of a control component (PWR and BWR)

- ~~Single failure~~Failure in the electrical system (PWR and BWR)

- Minor reactor coolant system (RCS) leak or loss of reactor coolant such as from a ~~small~~ ruptured small pipe or from a crack in a large pipe (PWR and BWR)

- Minor secondary system break (PWR)

- Loss of offsite power (PWR and BWR)

- Operation with a fuel assembly in an improper position (PWR and BWR)

- Inadvertent ~~blowdown of RCS~~ (opening of a pressure relief valve (PWR and BWR)

- Loss of feedwater heating (PWR and BWR)

- Trip of any or all recirculation pumps (BWR)

- Inadvertent pump start in a hot recirculation loop (BWR)

- Condenser tube leak (BWR)

- Startup of an idle recirculation pump in a cold loop (BWR)

- ~~Reactor overpressure with delayed scram~~

~~The individual event sections of the SRP address specific AOOs and their appropriate variations (e.g., design-specific variations).~~

~~Anticipated transients without scram (ATWSs) are AOOs in which a reactor scram is demanded but fails to occur because of a common-mode failure in the reactor scram system. ATWS events, therefore, are AOOs that postulate complete failure of the required (single-failure proof) protection system. As such, they are beyond the design basis, and consequently, ATWS events are addressed separately (see SRP Section 15.8).~~

Postulated accidents are unanticipated occurrences (i.e., they are postulated but not expected to occur during the life of the nuclear power plant). Note that some licensing basis documents may use the term "limiting fault" consistent with RG 1.70.

~~Postulated accidents are also known as Condition IV events in the unofficial ANS~~

~~standards.~~

The following are ~~some~~ examples of postulated accidents ~~in PWRs~~considered in previous licensing reviews for currently operating PWR and ~~BWRs of current~~BWR designs:

- ~~Major rupture~~Rupture of a large pipe containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant pressure boundary (PWR and BWR)
- Ejection of a control rod assembly (PWR)
- Control rod drop accident (BWR)
- Major secondary system pipe rupture up to and including double-ended rupture (PWR and BWR)
- Single reactor coolant pump locked rotor (PWR)
- Seizure of one recirculation pump (BWR)
- Steam generator tube rupture (PWR)

Fuel ~~The sections of the SRP dealing with the individual events address specific postulated accidents and appropriate variations (e.g., design-specific variations).~~

- handling accident or cask drop (PWR and BWR)

Finally, insights from operating experience have prompted regulatory actions to address a limited set of events beyond the scope of the design-basis criteria used to evaluate reactor designs. These beyond-design-basis events involve conditions not fully considered in the design process (e.g., the occurrence of multiple, independent failures) because they were judged to be too unlikely. Considering their very low likelihood of occurrence, beyond-design-basis events are typically assessed with best-estimate inputs in lieu of the conservative design-basis assumptions associated with design-basis events. The following are examples of beyond-design-basis events required to be addressed:

- Anticipated transient without scram (ATWS) (10 CFR 50.62)
- Station blackout (10 CFR 50.63)
- Loss of all alternating current (ac) power (10 CFR 50.155)

The individual subsections of Chapter 15 of the SRP provide guidance for the review of these specific events, except for loss of all ac power events.[1] For new reactor designs, the relevant set of analyzed events and their categorization may differ from the examples above.

B. Categorization According to Type. AOOs and postulated accidents are also categorized according to type. The type of AOO or postulated accident is defined by its effect on the plant. For example, one type of AOO or postulated accident will cause the RCS to pressurize and possibly jeopardize RCS integrity. Another type will cause the RCS to depressurize and possibly jeopardize fuel cladding

[1] RG 1.226 provides guidance on methods that the staff considers acceptable for use in implementing 10 CFR 50.155 (Ref. 14).

integrity. It is useful to categorize and organize analyses of AOOs and postulated accidents according to type, so that analysts can compare them on common bases, effects, and safety limits. Such comparisons can help to identify limiting events and cases for detailed examination and eliminate nonlimiting cases from further consideration.

AOOs and postulated accidents ~~can be~~considered in previous licensing reviews for currently operating PWR and BWR designs have been grouped into the following seven types:

(1) Increase in heat removal ~~by~~from the ~~secondary~~primary system
(2) Decrease in heat removal ~~by~~from the ~~secondary~~primary system
(3) Decrease in RCS flow rate
(4) Reactivity and power distribution anomalies
(5) Increase in reactor coolant inventory
(6) Decrease in reactor coolant inventory
(7) Radioactive release from a subsystem or component

~~The~~The reviewer ensures that all relevant accident types for the reactor design under review have been identified and that all potentially limiting events within these categories have been included in the analysis.

The reviewer ensures that the analysis of ~~AOOs~~each AOO and postulated accident ~~analyses, within~~ encompasses a ~~type, can (and should) encompass a~~ sufficient variety of cases or event scenarios, each designed to produce effects or results that challenge designated ~~safety limits.~~acceptance criteria. Depending upon the postulated initial conditions and equipment behavior, a given event may have the potential to challenge multiple acceptance criteria. In some cases, the same assumed initial conditions and equipment behavior will produce limiting conditions for multiple figures of merit. In other cases, different figures of merit will be challenged by different event scenarios. For example, one ~~case study of the~~ set of initial conditions and equipment performance assumptions could result in a turbine trip event~~, an AOO that causes a decrease in heat removal by the secondary system, can be designed to yield~~ that yields a high peak RCS pressure, and ~~another case study of~~ a different set of assumed conditions for the same AOO ~~can be designed to yield a low,~~could result in minimum thermal margin. The former case tests the ~~safety limit~~acceptance criterion for RCS pressure boundary integrity, while the latter case tests the safety limit that protects fuel cladding integrity.

The reviewer considers the possible case variations of AOOs, and postulated accidents presented to verify that the applicant or licensee has identified the limiting cases. The reviewer evaluates applicants' or licensees' claims that individual AOOs, and postulated accidents are limiting or nonlimiting, or bounded by other AOOs and postulated accidents, with particular attention to the bases used for comparison.

~~Comparison of AOOs to other AOOs within a type, for example, is easily justified. Comparison of AOOs of one type to postulated accidents~~ For example, in some cases, qualitative comparison of an AOO of one type to another AOO of the same type may be readily justified (e.g., all else being equal, an event involving a valve with a faster closure time may result in a larger peak pressure). On the other hand, considering the potential for complex phenomenological and system interactions, a qualitative disposition that an AOO of one type bounds a postulated accident of another type requires closer scrutiny and more justification from the licensee.

2. Analysis Acceptance Criteria. ~~If~~ The NRC's acceptance criteria for safety analysis recognize that the risk of an event is ~~defined as the product of~~derived from both the event's frequency of occurrence and the severity of its consequences~~, then the~~. The design of ~~the~~a plant should be such that all the AOOs and postulated accidents produce about the same acceptably low level of risk (i.e., the risk is approximately constant across the spectrum of AOOs and postulated accidents). This approach is reflected in the ~~general design criteria (~~GDC~~),~~, which generally ~~prohibit~~require plant design to preclude relatively frequent events (AOOs) from resulting in serious consequences~~, but allow the~~. For relatively rare events (postulated accidents~~) to produce more severe~~), the GDC achieve this acceptably low level of risk by requiring mitigation of potentially serious consequences using less restrictive acceptance criteria than those applicable to AOOs. For example, with respect to fuel performance, the fuel cladding fission product barrier must be maintained for all AOOs; however, for initiating events categorized as postulated accidents, a loss of fuel cladding integrity may be permitted under some circumstances and instead the fuel is required to remain amenable to cooling.

The reviewer will consider the results of ~~licensees'~~the applicant's or licensee's analyses and evaluations of individual ~~initiating~~ events to ascertain whether the applicant or licensee has satisfied the applicable analysis acceptance criteria for each of the events. The applicant or licensee may propose the use of alternate acceptance criteria appropriate to the particular plant design and operation (e.g., for new reactor design applications). In such cases, the reviewer will consider the alternate criteria and determine whether they are equivalent or more conservative, in function and consequences, to the current criteria ~~(see~~. A list of the fundamental criteria necessary to meet the requirements of GDCs for AOOs and postulated accidents is provided in Table 15.0-1 below~~)~~. Individual sections of the SRP may specify additional criteria pertaining to a certain postulated event.

~~A. Analysis Acceptance Criteria for AOOs. The following are the specific criteria necessary to meet the requirements of GDC for AOOs:~~

~~i. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with~~

~~the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.~~

Table 15.0-1: Analysis Acceptance Criteria		
Figure of Merit	**AOO**	**Postulated Accident**[2]
Fuel Performance	Specified acceptable fuel design limits (SAFDLs)[3]	Fuel damage shall be limited to the extent consistent with applicable radiological criteria achieve and maintain a safe and stable condition, and maintain the core in a known configuration amenable to cooling
Maximum RCS and Main Steam Pressures	ASME Service Level B	ASME Service Level C
Radiological Release	10 CFR Part 20	Graded approach of 10 CFR 50.34(a)(1), or equivalent[4]
Escalation Criteria	An AOO must not generate a postulated accident without other faults occurring independently or result in a loss of function of the RCS or reactor containment barriers.	A postulated accident must not, by itself, cause a loss of credited functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

Consistent with the policies and resolution of key technical issues for passive reactor designs (Refs. 10-13), the reviewer must confirm that advanced reactors that rely on passive safety-related systems and equipment for AOO and accident mitigation are designed with sufficient capability and capacity to reach and maintain safe and stable conditions without operator action and without non-safety-related onsite or offsite power for at least 72 hours. Further, the site must be capable of sustaining all design-basis events with onsite equipment and supplies for 7 days.

The reviewer should ensure the applicant's or licensee's analysis demonstrates the above acceptable criteria are satisfied for the first 72 hours following the event and the reactor remains subcritical beyond the short-term.[5] Additionally, the events should be evaluated into the long term to demonstrate the reactor will remain in a safe and stable condition beyond 72 hours. The reviewer should ensure there remain no degrading or transitory phenomena, and all figures of merit have a favorable trend.

[2] For loss-of-coolant accidents (LOCAs), additional acceptance criteria of 10 CFR 50.46(b) shall be satisfied.
[3] Typical SAFDLs for demonstrating fuel integrity include limits on the departure from nucleate boiling ratio (DNBR) or critical power ratio (CPR) to prevent overheating, fuel centerline temperature to prevent melting, and transient cladding strain to avoid cladding failure.
[4] For operating plants, examples of similar regulatory requirements include 10 CFR Part 100 and 10 CFR 50.67. See SRP 15.0.3 as an example of appropriate dose criteria for postulated accidents where the acceptance criteria for a given event are provided as fractions of the regulatory criteria.
[5] For an example, see SECY-18-0099, "NuScale Power Exemption Request from 10 CFR Part 50, Appendix A, General Design Criterion 27, 'Combined Reactivity Control Systems Capability'" (ADAMS Accession No. ML18065A540).

Transition to recovery or long-term support from non-safety-related systems, structures, and components (SSCs) is outside the scope of the staff's Chapter 15 review. Post-72-hour operator actions, defense-in-depth, and regulatory treatment of non-safety systems (RTNSS) is evaluated as part of the staff's review of Chapter 19 or corresponding system section.

~~ii. 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 and that the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs.~~

~~The reviewer applies a third criterion, based on the ANS standards to ensure that there is no possibility of initiating a postulated accident with the frequency of occurrence of an AOO. Some of the questions that licensees must answer to justify making plant modifications without advance review (see 10 CFR 50.59) by the NRC staff reflect this concern.~~

~~iii. An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.~~

~~For licensees that have the categorizations of References 4 or 5 (i.e., ANS Condition II, III, and IV events) in their licensing bases, the reviewer will apply the following acceptance criteria:~~

~~(1) Condition II events~~

~~(a) Same as Criterion (1) (above), for AOOs.~~

~~(b) Same as Criterion (2) (above), for AOOs.~~

~~(c) 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.~~

~~(2) Condition III events~~

~~(a) No more than a small fraction of the fuel elements in the reactor are damaged, although sufficient fuel element damage might occur to preclude resumption of operation for a considerable outage time.~~

~~(b) For PWRs, the release of radioactive material may exceed guidelines of 10 CFR Part 20, but shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius.~~

~~For BWRs, the offsite release of radioactive material is limited to a small fraction of the guidelines of 10 CFR Part 100, which may be the result of the failure of a small fraction of the fuel elements in the reactor.~~

~~(c) A Condition III incident shall not, by itself, generate a Condition IV fault or result in a consequential loss of function of the RCS or reactor containment barriers.~~

~~(3) Condition IV events~~

~~ANS Condition IV events are postulated accidents. The reviewer will apply the acceptance criteria for postulated accidents (below) to evaluate Condition IV events.~~

~~B. Analysis Acceptance Criteria for Postulated Accidents. Unlike an AOO, a postulated accident could result in sufficient damage to preclude resumption of plant operation. A list of the basic criteria necessary to meet the requirements of GDC for postulated accidents appears below. Individual sections of the SRP may specify additional criteria pertaining to certain postulated accidents.~~

~~i. Pressure in the RCS and main steam system should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.~~

~~ii. Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs. If the minimum DNBR or MCPR does not meet these limits, then the fuel is assumed to have failed.~~

~~iii. The release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR Part 100.~~

~~iv. A postulated accident shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.~~

~~For loss-of-coolant accidents (LOCAs), the following analysis acceptance criteria of 10 CFR 50.46 also apply:~~

~~i. The calculated maximum fuel element cladding temperature shall not exceed 2200 EF.~~

~~ii. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.~~

~~iii. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.~~

~~iv. Calculated changes in core geometry shall be such that the core remains amenable to cooling.~~

~~v. After any calculated successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature shall should be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.~~

3. Plant Characteristics Considered in the Safety ~~Evaluation~~Analysis. The reviewer ensures that the application contains the key plant parameters considered in the safety ~~evaluation~~analysis (e.g., core power, core inlet temperature, reactor system pressure, core flow, axial and radial power distribution, fuel and moderator temperature coefficient, void coefficient, reactor kinetics parameters, available shutdown rod worth, and control rod insertion characteristics). The reviewer checks that the range of values for plant parameters is ~~representative of fuel exposure or core reload, and that the range is~~acceptable, consistent with technical specifications, and sufficiently broad to cover the predicted fuel cycle ranges, to the extent practicable, based on the fuel design and acceptable analytical methodology at the time of the ~~LAR, DC, or COL~~ application.

The reviewer also ensures that the application specifies the permitted fluctuations and uncertainties associated with reactor system parameters and assumes the appropriate conditions, within the operating band, as initial conditions for transient analysis.

4. Assumed Protection and Safety Systems Actions. The reviewer ensures that the application lists the settings of all the protection and safety systems functions that are used (i.e., credited) in the safety ~~evaluation~~analysis. Typical protection and safety systems functions include reactor trips, isolation valve closures, emergency core cooling system (ECCS) and decay heat removal system initiation and ~~ECCS~~performance. In evaluations of AOOs and postulated accidents, the performance of each credited protection or safety system is required to include the effects of the most limiting postulated single ~~active~~ failure. This verifies satisfaction of the GDC ~~criteria~~ that require protection and safety systems to adequately perform their intended safety functions in the presence of single ~~active~~ failures. The reviewer also ascertains that the application lists the expected limiting delay time for each protection or safety system function and describes the ~~acceptable~~ methodology for determining uncertainties (from the combined effects of calibration error, drift, instrumentation error, and other factors) to be included in the establishment of the trip setpoints and allowable values specified in the plant technical specifications. Review of the instrumentation setpoint methodology is performed in accordance with Chapter 7 of the SRP.

5. Evaluation of Individual ~~Initiating~~ Events. The reviewer ensures that the application includes an evaluation of each ~~initiating~~analyzed event, using the format in Subsection I.6 of this SRP section. For ~~initiating~~ events that are determined to be not limiting, the reviewer may evaluate qualitative justifications and conduct comparisons with ~~the corresponding,~~ more limiting ~~initiating~~ events in the same event category.

6. Event Evaluation

   A. Identification of Causes and Frequency Classification. For each initiating event evaluated, the reviewer ensures that the application includes a description of the occurrences that can lead to the event and a categorization of the event as either an AOO or postulated accident. The reviewer also checks for clear definitions of the analysis acceptance criteria appropriate to the specific nature of the initiating event, as well as the event’s categorization.

   B. Sequence of Events and Systems Operation. The reviewer verifies that the application addresses the following considerations for each ~~initiating~~ event:

      i. Step-by-step sequence of events, from event initiation to the final stabilized condition (i.e., identification on a time scale of each significant occurrence, including flux monitor trips, insertion of control rods, attainment of primary coolant safety valve ~~set points~~setpoints, opening and closing of safety valves, generation of containment isolation signals, and containment isolation) and identification of all operator actions credited in the transient and accident analyses for consequence mitigation

      ii. Extent to which normally operating plant instrumentation and controls are assumed to function

iii. Extent to which plant and reactor protection systems are required to function

iv. Credit taken for the functioning of normally operating plant systems

v. Credited operation of engineered safety systems

vi. ~~Assurance of~~If applicable and available,[6] consistency between the safety analyses and the emergency response guidelines/emergency procedure guidelines or emergency operating procedures with respect to the operator response (including action time) and available instrumentation

The reviewer verifies that the applicant has specified only safety-related systems or components for use in mitigating AOO and postulated accident conditions~~,~~ and has included the effects of single ~~active~~ failures in those systems and components. ~~The reviewer may consider~~In the ~~licensee's technical justifications~~event that applicants or licensees propose an exception to allow credit for the operation of ~~nonsafety~~certain non-safety-related ~~systems or~~ equipment, the reviewer should review the technical justification and ensure the safety analysis does not credit the non-safety-related SSC as a frontline mitigating system or as being within the primary success path necessary for satisfying the acceptance criteria in Table 15.0-1. For instance, NUREG-0138 (Ref. 9), Issue 1, discusses credit for non-safety-related components ~~(e.g., when they are used as~~ in the main steam and feedwater systems as a backup protection ~~and when they are not disabled, except by a detectable, random, and independent~~when the single failure~~).~~ criterion is applied to the analysis of a postulated main steam line break. NUREG-0138 contains guidance on considerations for component type and reliability, and technical specification surveillance requirements.

Considering the set of permissible equipment operating configurations applicable to each event, the reviewer verifies that the applicant has analyzed the performance of non-safety-related systems or components in the operating condition that leads to the most limiting results. For instance, the reviewer should verify that the analysis considers non-safety-related systems normally in operation at the time of the event as continuing to function; such systems should not be assumed to experience a random and independent failure that would result in the prevention or termination of the event (e.g., assumed continued operation of the main feedwater control system for the analysis of inadvertent control rod withdrawal event). Non-safety-related systems that are in a standby mode during normal operation should not be considered a primary success path or support a mitigating system in the safety analysis.

For LAR, OL, DC, and COL reviews, in accordance with Criterion 3 specified in 10 CFR 50.36(c)(2)(ii)(C), the technical specifications must include limiting conditions of operation (LCOs) for equipment credited in the transient and accident analyses.

The reviewer ascertains that the applicant has evaluated the effects of the most

[6] Emergency response guidelines or emergency procedure guidelines may not be available during initial licensing review. For passive reactor designs, such guidelines are not applicable because operator actions should not be credited in the Chapter 15 safety analysis.

limiting single ~~active~~ failures and operator errors and that the applicant's or licensee's application contains sufficient detail to permit independent evaluation of the adequacy of systems, as they relate to the subject events. A single failure is an occurrence which results in the loss of capability of a component to perform its intended safety functions.[7]

For single failures in electrical and instrumentation and control (I&C) components, both passive and active failures should be considered in addition to the initiating event. The failure may be postulated to occur at any time during the event.

An active failure in a fluid system means the failure of a component that relies on mechanical movement for its operation to complete its intended function on demand, or an unintended movement of the component. Examples of components that require mechanical movement include air-operated valves, check valves, and pumps. A passive failure in a fluid system means a breach in the fluid pressure boundary or a mechanical failure which adversely affects a flow path. Passive failures in fluid systems should be considered as initiating events and during long-term core cooling modes of operation (e.g., degradation and leakage from valve or pump seals). For additional guidance related to applying the single failure criterion to systems and components see SECY-77-439 (Ref. 6), and regarding the current Commission position on the treatment of check valves in passive safety systems see SECY-94-084 (Ref. 11).

C. Core, System, and Barrier Performance

i. Evaluation Model. The reviewer ensures that the applicant has discussed the evaluation model used and any simplifications or approximations introduced to perform the analyses and identified digital computer codes used in the analysis. If the analysis uses more than one computer code, the applicant should describe the method used to connect the codes. ~~The reviewer verifies that the applicant has discussed the important output of the codes under "results" with emphasis on the input data and the extent or range of variables investigated and~~ The reviewer verifies that the applicant has included detailed descriptions of evaluation models and digital computer codes or listings by referencing documents that are available to the NRC.

The reviewer ensures that the applicant has provided a ~~table~~ listing of ~~the titles of~~ topical reports (TRs) that describe models or computer codes used in transient and accident analyses and listed the associated NRC safety ~~evaluation reports~~evaluations (SEs) approving those TRs. The reviewer checks that implementations of NRC-approved models or codes are within the applicable ranges and conditions and that the applicant has demonstrated compliance with each of the conditions and limitations imposed by the NRC staff in its ~~safety evaluation reports~~SEs that approve the TRs.

[7] Appendix A to 10 CFR Part 50, "Definitions and Explanations," gives a more complete definition of "single failure."

If the applicant or licensee is referencing evaluation models, analytic methods, or digital computer codes that have not been approved for use by the NRC for the intended application, then the reviewer should confirm that the application describes the models, methods, and codes and provides information to justify their use in accordance with SRP 15.0.2 and RG 1.203 (Ref. 15).

~~ii.~~ Input Parameters and Initial Conditions. The reviewer verifies that the applicant has: (1) identified the major input parameters and initial conditions used in the analyses; (2) included the initial values of other variables and parameters in the application if they are used in the analyses of the particular event under study; (3) ensured that the parameters and initial conditions used in the analyses are suitably

ii. conservative[8]; and (4) discussed the bases (including the degree of conservatism) used to select the numerical values of the input parameters. The reviewer verifies that the input parameters and initial conditions are consistent with applicable technical specifications.

iii. Results. The reviewer ensures that the applicant has presented the results of the analyses, including key parameters as a function of time during the course of the transient or accident. The following are examples of parameters that should be included:

- Neutron power
- Thermal power
- Heat fluxes, average and maximum
- RCS pressure
- DNBR or CPR, as applicable
- Core and recirculation loop coolant flow rates for BWRs
- Coolant conditions, including inlet temperature, core average temperature (for PWRs), core average steam volume fraction (for BWRs), average exit and hot channel exit temperatures, and steam volume fractions
- Temperatures, including maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy
- Reactor coolant inventory, including total inventory, and coolant level in various locations in the RCS
- Secondary (power conversion) system parameters, including steam flow rate, steam pressure and temperature, feedwater flow rate, feedwater temperature, and steam generator inventory
- ECCS flow rates and pressure differentials across the core, as applicable
- Containment pressure
- Relief and/or safety valve flow rate
- Flow rate from the RCS to the containment system, if applicable
- Pressurizer water volume (for PWRs)

[8] "Suitably conservative" means the input values used for parameters and initial conditions represent either 1) an appropriately conservative set of possible conditions, or 2) realistic conditions in concert with an appropriate treatment of the associated uncertainties and variabilities.

In addition, the discussion of the results should emphasize the margins between the predicted values of various core parameters, as well as the values of those parameters that would represent limiting acceptable conditions.

~~Review Interfaces~~

~~Other SRP sections interface with this section as follows:~~

~~1. Design basis radiological consequence analyses associated with design basis accidents are reviewed under SRP Section 15.0.3.~~

~~The specific acceptance criteria and review procedures are contained in the referenced SRP section.~~

II. ACCEPTANCE CRITERIA

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

1. 10 CFR Part 20, "Standards for Protection Against Radiation"

2. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (especially 10 CFR 50.46 and Appendix A)

3. 10 CFR Part 100, "Reactor Site Criteria"

4. 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants"

The following GDC from Appendix A to 10 CFR Part 50 are relevant to SRP Section 15:

~~1. GDC 2, as it relates to the seismic design of structures, systems, and components (SSCs) whose failure could cause an unacceptable reduction in the capability of the residual heat removal system.~~

~~2. GDC 4, as it relates to the requirement that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions, including such effects as pipe whip and jet impingement.~~

~~3. GDC 5, as it relates to the requirement that any sharing among nuclear power units of SSCs important to safety will not significantly impair their safety function.~~

~~4.~~1. GDC 10, as it relates to the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations including AOOs.

2. GDC 12, as it relates to the reactor core and associated coolant, control, and protection systems being designed to assure that power oscillations are not possible or can be reliably and readily detected and suppressed.

~~5.~~3. GDC 13, as it relates to instrumentation and controls provided to monitor variables over anticipated ranges for normal operations, for AOOs, and for accident conditions.

4. GDC 14, as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal

leakage, of rapidly propagating failure, and of gross rupture.

~~6.~~5. GDC 15, as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs.

6. GDC 16, as it relates to containment and associated systems to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and containment design conditions important to safety not being exceeded for as long as postulated accident conditions require.

7. GDC 17, as it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of SSCs important to safety. The safety function for each system (assuming the other system is not working) shall be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during an AOO and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.

8. GDC 19, as it relates to the requirement that a control room be provided from which personnel can operate the nuclear power unit during both normal operating and accident conditions, including a LOCA.

9. GDC 20, as it relates to the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the plant does not exceed specified acceptable fuel design limits during any condition of normal operation, including AOOs.

10. GDC 25, as it relates to the requirement that the reactor protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods.

11. GDC 26, as it relates to the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded even during AOOs. This is accomplished by ensuring that the applicant has allowed an appropriate margin for malfunctions such as stuck rods.

12. GDC 27 and 28, as they relate to the RCS being designed with an appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.

13. GDC 29, as it relates to the design of the protection and reactivity control systems and their performance (i.e., to accomplish their intended safety functions) during AOOs.

14. GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner, and that the probability of propagating fracture is minimized.

15. GDC 34, as it relates to the capability to transfer decay heat and other residual heat from the reactor so that fuel and pressure boundary design limits are not exceeded.

16. GDC 35, as it relates to the RCS and associated auxiliaries being designed to provide abundant emergency core cooling.

17. GDC 38, as it relates to a system to remove heat from containment to rapidly reduce pressure and temperature, consistent with the functioning of other systems, following any loss-of-coolant accident and maintain them at acceptably low levels.

18. GDC 50, as it relates to the containment and its heat removal system being designed to accommodate the calculated pressure and temperature conditions resulting from any loss-of-coolant accident without exceeding the design leakage rate and with sufficient margin based on consideration of several factors.

~~17.~~19. GDC 55, as it relates to the isolation requirements of small-diameter lines connected to the primary system.

~~18.~~20. GDC 60, as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment.

~~19.~~21. GDC 61, as it relates to the requirement that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

Subsection I.2 of this SRP section discusses general acceptance criteria, and SRP Chapter 15 subsections discuss specific acceptance criteria for transients or accidents, as appropriate.

III. REVIEW PROCEDURES

The reviewer will select material from the procedures described below, as may be appropriate for a particular case.

These review procedures are based on the identified SRP 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.

To evaluate ~~the LAR, DC, or COL application~~standard design and license applications under 10 CFR Parts 50 and 52, the reviewer verifies that the applicant has performed the applicable transient and accident analyses needed to demonstrate conformance to the regulations.

SRP Chapter 15 subsections discuss specific review procedures for transients or accidents.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support the conclusions~~ of the following type~~ to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

SRP Chapter 15 subsections discuss the statements and conclusions of evaluation findings for transients or accidents.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of ~~DC~~ applications ~~and license applications~~for licensing actions submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an ~~acceptable~~ alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations. If an application proposes an alternative to a method described herein, the staff will evaluate whether the alternative demonstrates compliance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted six months or more after the date of issuance of this SRP section.

The referenced regulatory guides contain implementation schedules for conformance to parts of the ~~method~~methods discussed ~~here~~herein.

VI. DEFINITIONS

Term	Definition
anticipated operational occurrences (AOOs)	Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power. ~~AOOs are also known as Condition II and III events.~~
anticipated transient without scram (ATWS)	A beyond-design-basis event consisting of an AOO followed by the failure of the reactor trip portion of the protection system specified in GDC 20~~, because of common-mode failure~~.
~~common-mode failure~~	~~The result of an event which, because of dependencies, causes a coincidence of failure states of components in two or more separate channels of a redundancy system, leading to the failure of the defined system to perform its intended function.~~
critical power ratio (CPR)	~~That~~The power in ~~the~~a fuel assembly that will cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
~~departure from nucleate boiling (DNB)~~	~~The DNB acceptance criterion for an AOO is met when there is a 95 percent probability at a 95 percent confidence level (the 95/95 DNB criterion) that DNB will not occur, and the fuel centerline temperature stays below the melting temperature.~~
beyond-design-~~-~~basis event	A possible accident sequence not fully considered in the reactor design process because it was judged to be too unlikely. When such events are analyzed, analysis is typically performed with best-estimate inputs and assumptions in lieu of conservative design-basis modeling approaches.~~Information that identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design.~~ ~~These values may be (1) restraints derived from generally accepted state of the art practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.~~
~~design-basis accidents~~	~~Postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components.~~
~~design-basis events~~departure from nucleate boiling ratio (DNBR)	~~Conditions of normal operation, including AOOs, design-basis accidents, external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and~~

	~~maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures.~~The heat flux that will cause some point on a fuel rod to experience a rapid decrease in heat transfer due to the insulating effect of a vapor blanket that forms on the rod surface, divided by the actual local heat flux.

DEFINITIONS (CONTINUED)

Term	Definition
design-basis	Information that identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted state of the art practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.
~~general~~ design ~~criteria (GDC)~~-basis accidents	~~Reference 1 lists the GDC. The GDC that mention AOOs are 10, 13, 15, 17, 20, 26, 29, 60, and 64. The GDC that mention postulated accidents are 4, 16, 17, 22, 27, 28, 31, 41, 51, 61, and 64.~~Postulated accidents that are used to set design criteria and limits for the design and sizing of systems and components important to safety, including safety-related SSCs.
design-basis events	Conditions of normal operation, including AOOs, design-basis accidents, external events, and natural phenomena, for which the plant must be designed and analyzed.
loss-of-coolant accident (LOCA)	A postulated accident that results in the loss of reactor coolant at a rate in excess of the replacement capability of the reactor coolant makeup system.
~~MCPR safety limit~~	~~This limit ensures that during normal operation and during AOOs, at least 99.9 percent of the fuel rods in the core do not experience transition boiling.~~
~~minimum critical power ratio (MCPR)~~	~~The smallest CPR that exists in the core for each class of fuel.~~
overpressurization	The condition that occurs when the pressure increase for a given event exceeds ~~the design~~an acceptance criterion established by acceptable means for maximum pressure ~~of the component of interest by more than 10 percent, in accordance~~associated with ~~the ASME Code~~that event.
postulated accidents	~~Unanticipated conditions~~Conditions of operation ~~(i.e.,~~ not expected to occur during the life of the nuclear power unit~~).~~ ~~Postulated accidents are also known as Condition IV events~~ for which a structure, system, or component must meet its functional goals.

protection system	~~The protection~~A system ~~shall be designed~~ to (1) ~~to initiate~~ automatically initiate the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) ~~to~~ sense accident conditions and to initiate the operation of systems and components important to safety. (GDC-20)
single failure	An occurrence that results in a component's loss of capability to perform its intended safety functions. Multiple consequential failures resulting from a single occurrence are considered to be a single failure.

VII. REFERENCES

1. Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Plants."

2. Regulatory Guide 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.~~"~~" (ADAMS Package Accession No. ML011340122).

3. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition~~)."~~)," June 2007 (ADAMS Package Accession No. ML070720184).

~~4. ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants" (replaces ANSI N18.2), 1983 (withdrawn in 1998).~~

~~5. ANSI/ANS-52.1-1978, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants" (withdrawn in 1998).~~

~~6.~~4. 10 CFR 50.62, “Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water Cooled Nuclear Power Plants.”

~~7.~~5. ~~ASME~~American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section~~-~~ III, “Nuclear Power Plant Components,” Article NB-7000, “Protection Against Overpressure~~,” American Society of Mechanical Engineers.~~.”

~~8.~~6. SECY-77-439, “Single-Failure Criterion,” August 1977 (ADAMS Accession No. ML060260236).

~~9.~~7. 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.”

~~10.~~8. 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”

~~PAPERWORK REDUCTION ACT STATEMENT~~

9. ~~The information collections contained~~NUREG-0138, “Staff Discussion of 15 Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director, NRR to NRR Staff,” November 1976 (ADAMS Accession No. ML13267A423).

10. 73 FR 60612, “Policy Statement on the Regulation of Advanced Reactors,” October 2008.

11. SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” March 1994 (ADMAS Accession No. ML003708068).

12. SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084),” May 1995 (ADAMS Accession No. ML003708005).

13. SECY-96-128, “Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design,” June 1996 (ADAMS Accession No. ML003708224).

14. Regulatory Guide 1.226, “Flexible Mitigation Strategies for Beyond-Design-Basis Events,” June 2019 (ADAMS Accession No. ML19058A012).

15. Regulatory Guide 1.203, “Transient and Accident Analysis Methods,” December 2005 (ADAMS Accession No. ML053500170).

Standard Review Plan ~~are covered by the requirements of~~ **Section 15.0, Draft Revision 4**
“Introduction - Transient and Accident Analyses”
Description of Changes

This Standard Review Plan (SRP) section affirms the technical accuracy and adequacy of the guidance previously provided in SRP Section 15.0, Revision 3, dated March 2007. See Agencywide Documents Access and Management System Accession No. ML070710376.

The main purpose of this revision is to update and reformat the analysis acceptance criteria consistent with the regulations. Specifically, outdated legacy information was removed and acceptance criteria were reformatted into tabular form. This update also provides additional staff guidance for new reactor-specific policies related to 72-hour and 7-day coping capabilities, safe and stable condition, and plant recovery following a design-basis event.

Section I.6 was updated to clarify staff guidance related to non-safety-related SSCs and single failure assumptions.

Additional references were added to provide context and support specific staff guidance or Commission positions.

Additional changes provided minor clarifications or were editorial in nature.

Paperwork Reduction Act

This Standard Review Plan provides voluntary guidance for implementing the mandatory information collections in 10 CFR ~~Part~~Parts 50 and ~~10 CFR Part~~ 52~~, and~~ that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget~~, approval number~~ (OMB), under control numbers 3150-0011 and 3150-0151~~.~~ respectively. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch ((T6 A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555 0001, or by email to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB 10202 (3150-0011 and 3150-0151), Office of Management and Budget, Washington, DC, 20503; email: oira_submission@omb.eop.gov.

~~PUBLIC PROTECTION NOTIFICATION~~

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a ~~request for~~collection of information~~ or an information collection requirement~~ unless the document requesting ~~document~~ or requiring the collection displays a currently valid OMB control number.