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(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.2.1-15.2.5 LOSS OF EXTERNAL LOAD; TURBINE TRIP; LOSS OF CONDENSER VACUUM; CLOSURE OF MAIN STEAM ISOLATION VALVE (BWR); AND STEAM PRESSURE REGULATOR FAILURE (CLOSED)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - None

I. AREAS OF REVIEW

A number of transients which are expected to occur with moderate frequency result in unplanned decreases in heat removal by the secondary system. Each transient covered in this SRP section should be discussed in individual sections of the safety analysis report (SAR), as required by the Standard Format (Ref. 1). The transients to be evaluated are:

1. Loss of External Load

In a loss of external load event an electrical disturbance causes loss of a significant portion of the generator load. This loss of load situation is different from the loss of ac power condition considered in Standard Review Plan (SRP) Section 15.2.6 in that offsite ac power remains available to operate the station auxiliaries (such as reactor coolant pumps). The onsite emergency diesels are therefore not required for the loss of external load transient. Immediate fast closure of the turbine control valves (TCV) and intercept valves is initiated whenever a loss of generator load takes place. For a boiling water reactor (BWR), a fast TCV closure (0.150-0.2 sec) causes a sudden reduction in steam flow and results in a reactor pressure surge. For a BWR without select rod insert (SRI), reactor scram occurs. For a pressurized water reactor (PWR) there is also a sudden reduction in steam flow, and this causes the pressure and temperature in the shell side of the steam generator to increase. The latter effect, in turn, results in an increase in reactor coolant temperature, a decrease in coolant density, an increase in water volume in the pressurizer, and an increase in reactor coolant pressure. For a PWR with an integrated control system, reactor power can be run back to a lower level on TCV closure.

In all light water-cooled reactors, sensible and decay heat can be removed through actuation of one or several of the following systems: steam relief system, steam bypass to the condenser, reactor core isolation cooling system (BWR), emergency core cooling systems, and auxiliary feedwater system (PWR).

2. Turbine Trip

In a turbine trip event a malfunction of a turbine or reactor system causes the turbine to be tripped off the line by abruptly stopping steam flow to the turbine. This is different from the loss of electrical load condition described above in that fast closure of the turbine stop valves (TSV) is initiated. The TSV have faster (0.1 sec) closure times that the turbine control valves, resulting in more severe transients. For typical BWR and PWR plants, position switches on the TSV sense the trip and initiate reactor scram. The remainder of this transient is similar to the previously discussed loss of electrical load.

3. Loss of Condenser Vacuum

A loss of condenser vacuum event is one of the malfunctions that can cause a turbine trip. The remarks in item 2, above, thus apply to this transient.

4. Main Steam Isolation Valve Closure

The main steam isolation valve (MSIV) transient for BWRs can be initiated by various steam line or reactor system malfunctions and by various operator actions. As the MSIVs close, position switches initiate a reactor scram when the valves in three or more of the steam lines are less than 90% open, the reactor pressure is above 600 psi, and the reactor mode switch is in the RUN position. The effect of MSIV closure is to limit steam flow to the turbine. The results are similar to those discussed in item 1, above, but tend to be less severe since the MSIV closure time is much longer than that of the TCV.

5. Steam Pressure Regulator Failure

Steam pressure regulator failure in a closed position yields a transient similar to the transients discussed above. Generally, because the rate of change of system parameters is slower for a steam pressure regulator failure, a less severe transient results.

The review of the transients described above includes the sequence of events, the analytical models, the values of parameters used in the analytical models, and the predicted consequences of the transients.

The sequence of events described in the SAR analysis is reviewed by RSB in consultation with the Instrumentation and Control Systems Branch (ICSB). The RSB reviewer concentrates on the assumptions used for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by RSB to ascertain that all mathematical models and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method or code has not been previously reviewed, the RSB reviewer requests initiation of a generic evaluation of the new analytical model or code by RSB or the Core Performance Branch (CPB), as appropriate.

The predicted results of the transient analyses are then reviewed to assure that the consequences meet the acceptance criteria given in subsection II, below. Further, the results of the analyses are reviewed to ascertain that the predicted values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

The RSB will coordinate other branches' evaluations that interface with the overall review of the transient analysis as follows: The ICSB review of SRP Sections 7.2 and 7.3 is consulted on the instrumentation and controls aspects of the sequences described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. ICSB also reviews potential bypass modes, and the possibility of manual control by the operator as part of its primary review responsibility for SRP Sections 7.2 through 7.5. The Core Performance Branch (CPB) reviews the values of all the parameters used in the analytical models, including the initial conditions of the core and system. CPB also reviews the core physics, fuel design, and core thermal-hydraulics data used in the SAR analysis as part of its primary review responsibility for SRP Sections 4.2 through 4.4. The review of the Technical Specifications is coordinated and performed by the Licensing Guidance Branch as part of its primary review responsibility for SRP Section 16.0.

For those areas of review identified above as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

RSB acceptance criteria are based on meeting the requirements of the following regulations:

A. General Design Criterion 10 as it relates to the reactor coolant system being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences.

B. General Design Criterion 15 as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to assure that the pressure boundary will not be breeched during normal operations including anticipated operational occurrences.

C. General Design Criterion 26 as it relates to the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, is accounted for.

Specific criteria necessary to meet the relevant requirements of GDC 10, 15, and 26 are as follows:

1. The basic objectives of the review of the transients listed in subsection I are:

   a. To identify which of the moderate-frequency transients that results in an unplanned decrease in secondary system heat removal is the most limiting. (The term "moderate frequency" is used in this SRP section in the same sense as in the definitions of design and plant process conditions in References 8 and 9.)

b. To verify that, for the most limiting transient, the predicted plant response is such that the specific criteria given below regarding fuel damage and system pressure are satisfied.

c. To verify that the plant protection systems setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies as delineated in Regulatory Guide 1.105 (Reference 3).

2. The criteria for incidents of moderate frequency are:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (Ref. 2).

b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failure must be assumed for all rods for which the DNBR or CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptale fuel damage model (see SRP Section 4.2) that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

3. The applicant should analyze these transients using an acceptable analytical model. The equations, sensitivity studies, and models described in References 4 through 7 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by RSB.

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

a. The reactor is initially at 102% of the rated (licensed) core thermal power (to account for a 2% power measurement uncertainty), and primary loop of flow is at the nominal design flow less the flow measurement uncertainty.

b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core for PWRs and a 0.8 multiplier on the predicted reactivity insertion rate for BWRs.

c. 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.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of these transients presented by the applicant in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

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. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods (see II.3.b) is accounted for.

If the SAR states that any one of these transients is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the reduction-of-heat-removal transient that is determined to be most limiting. For this transient, the RSB reviewer, in consultation with the ICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to adequately limit the consequences of the transient to an acceptable level. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The ICSB reviewer provides consultation on automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of systems and components which may affect the course of the transient. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7, and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, RSB initiates a generic review of the model proposed by the applicant. CPB is consulted regarding the specified acceptable fuel design limits (SAFDLs).

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

The results of the analysis are reviewed and compared to the acceptance criteria presented in subsection II of this SRP section regarding fuel integrity and the maximum pressure in the reactor coolant and main steam systems. The variations with time during the transient of the neutron power, heat fluxes (average and maximum), reactor coolant system pressure, minimum DNBR (PWR) or CPR (BWR); core and recirculation loop coolant flow rates (BWR), coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), 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 (if applicable) are reviewed. The more important of these parameters for the limiting transient are compared to those predicted for other similar plants to verify that they are within expected range.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report (SER):

> The staff concludes that the plant design is acceptable with regard to transients resulting in unplanned decreases in heat removal by the secondary system that are expected to occur with moderate frequency and that the predicted response meets the requirements of General Design Criteria 10, 15, and 26. This conclusion is based on the following:
>
> The applicant has met the requirements of GDC 10 and 26 with respect to demonstrating that specified acceptable fuel design limits are not exceeded for this event and has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure limits have not been exceeded by this event, and that resultant leakage will be within acceptable limits by assuring that plant transients do not result in an unplanned decrease in heat removal by the secondary system. Those that might be expected to occur with moderate frequency are turbine trip, loss of external load, steam pressure regulator malfunctions, main steam isolation valve closure (in BWRs), loss of condenser vacuum, loss of nonemergency ac power to the station auxiliaries, and loss of normal feedwater flow.* All these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _______ transient. This transient was

*The SER should present one statement for moderate frequency transients involving an unplanned decrease in heat removal by the secondary system. Thus, the results of the reviews under SRP Sections 15.2.6 and 15.2.7 are included in this statement.

evaluated by the applicant using a mathematical model that had 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 and in accordance with the recommendation of Regulatory Guide 1.105. The results of the analysis of the ____________ transient showed that cladding integrity was maintained by ensuring that the maximum departure from nucleate boiling ratio (or minimum critical heat ratio for a BWR) did not decrease below _____, and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of their design pressures.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides.

VI. REFERENCES

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

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

3. Regulatory Guide 1.105, "Instrument Spans and Setpoints."

4. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," General Electric Reports NEDO-24154 and NEDE-24154P Volumes I, II, and III, October 1978.

5. "Loftran Code Description," Westinghouse Electric Corporation Report WCAP-7907, October 1972 (in review).

6. "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, April 1974 (in review).

7. "TRAP2-Fortran Program for Digital Simulation of the Transient Behavior of the Once-Through Steam Generator and Associated Reactor Coolant Systems," Babcock & Wilcox, BAW-10128, August 1976 (in review).

8. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

9. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

10. 10 CFR Part 50, General Design Criterion 10, "Reactor Design."

11. 10 CFR Part 50, General Design Criterion 15, "Reactor Coolant System Design."

12. 10 CFR Part 50, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."