

Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI-2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radioisotopes within the containment building.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI-2 accident can be adequately met without reliance on a PASS.

Therefore, this change does not involve a significant reduction in the margin of safety.

The NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: April 3, 2003.

Description of amendment request: The proposed changes will revise the Updated Final Safety Analysis Report to change the Reactor Vessel Material Surveillance Program. The change will reflect participation in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP).

Basis for proposed no significant hazards consideration determination: As required by Section 50.91(a) of Title 10 of the Code of Federal Regulations (10 CFR), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

Changes in the fracture toughness properties of reactor vessel beltline materials, resulting from the neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance with the requirements of 10CFR50, Appendix H. The proposed change implements an integrated surveillance program that has been evaluated by the NRC [U.S. Nuclear Regulatory Commission] staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50. The BWRVIP's ISP surveillance material selection process adequately ensures that materials in the program effectively provide meaningful information to monitor changes in fracture toughness for GGNS [Grand Gulf Nuclear Station, Unit 1, or Grand Gulf] RPV [Reactor Pressure Vessel] materials. In addition, the ISP program requires participants to acquire and evaluate relevant ISP test data from the program which may affect RPV integrity evaluations in a timely manner. One advantage of participating in the BWRVIP ISP is that surveillance test data applicable to the Grand Gulf RPV will be available sooner than under the current plant specific program.

The proposed change will not affect current RPV performance and will not cause the RPV or interfacing systems to be operated outside of their design or testing limits. The

proposed change will not alter any assumptions previously made in evaluating the radiological consequences of accidents.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The change does not affect the design, function, reliability, or operation of any plant structure, system or component. The purpose of the reactor vessel material surveillance program is to monitor neutron embrittlement and thermal environment effects in order to predict the behavioral characteristics of materials of pressure retaining components of the reactor coolant pressure boundary and to ensure that reactor vessel fracture toughness and integrity requirements are not violated. The ISP is an approved alternate monitoring program that meets the regulatory requirements in Appendix H to 10 CFR 50. As an acceptable alternate monitoring program, the ISP cannot create a new failure mode involving the possibility of a new or different kind of accident.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from that previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The reactor material surveillance program required by 10 CFR 50, Appendix H, is designed to ensure that adequate margins of safety are provided for the reactor coolant pressure boundary during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. Monitoring changes in the fracture toughness of reactor vessel materials ensures that material changes due to radiation embrittlement are adequately considered for safe reactor operations. Paragraph III.C of Appendix H to 10 CFR 50 delineates the regulatory requirements for an ISP. The BWRVIP ISP meets these requirements and has been approved by the NRC.

One of the uses of the material surveillance data obtained through the proposed ISP is to ensure the reactor coolant system P/T [Pressure/Temperature] limits established by the Technical Specifications are conservative. The material surveillance data obtained through the proposed Integrated Surveillance Program will provide new information that will be evaluated to ensure that the P/T limits are conservative. In addition, a neutron fluence calculation methodology which has been approved by the NRC staff and is consistent with the attributes identified in U.S. Nuclear Regulatory Commission Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," will be used for the determination of reactor vessel and surveillance capsule neutron fluence values to ensure quality of the method and compatibility between ISP results.