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CONSEQUENCE EVALUATION OF HYPOTHETICAL
REACTOR PRESSURE VESSEL SUPPORT FAILURE*

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ABSTRACT

This paper describes a consequence evaluation to address safety concerns raised by the radiation embrittlement of the reactor pressure vessel (RPV) supports for the Trojan nuclear power plant. The study comprises a structural evaluation and an effects evaluation and assumes that all four reactor vessel supports have completely lost the load carrying capability. The structural evaluation concludes that the Trojan reactor coolant loop (RCL) piping is capable of transferring loads to the steam generator (SG) supports and the reactor coolant pump (RCP) supports and that the SG supports and the RCP supports have sufficient design margins to accommodate additional loads transferred to them through the RCL piping. The effects evaluation, employing a systems analysis approach, investigates initiating events and the reliability of the engineered safeguard systems as the RPV is subject to movements caused by the RPV support failure. The evaluation identifies a number of areas for further investigation and concludes that a hypothetical failure of the Trojan RPV supports due to radiation embrittlement will not result in consequences of significant safety concerns.

1 INTRODUCTION

The reactor pressure vessel (RPV) support embrittlement problem associated with pressurized water reactors (PWRs) in nuclear power plants (Cheverton et al. 1988) is that low-temperature irradiation of structural materials can result in RPV support structure embrittlement, increasing the potential for unstable propagation of flaws that might exist in the materials. The radiation-induced embrittlement may result in failure of the RPV supports and consequent movements of the reactor vessel, given the occurrence of a transient stress or shock such as could be experienced in a loss-of-coolant- accident (LOCA) or severe earthquake. The objective of this consequence evaluation is to provide a sound technical basis for determining whether the failure of reactor pressure vessel supports could prevent safe shutdown or lead to unacceptable consequences during or following the design basis earthquake or pipe rupture.

The evaluation is divided into two phases. Phase 1 is a pilot study on a selected nuclear plant. Phase 2 is a parametric study of critical variables undertaken in an attempt to generalize the pilot results to other nuclear units susceptible to neutron embrittlement damage. The Trojan nuclear power plant has been selected for the pilot study because its RPV supports are located in the high radiation zone and are subject to high tensile stresses. This paper summarizes the results and conclusion of the Phase 1 pilot study.

The pilot study comprises a structural evaluation and an effects evaluation for postulated failure of one or more RPV supports. As a bounding case in the Phase 1 study, all four supports of the Trojan reactor pressure vessel are assumed to have initially failed. Failure of a RPV support herein means the support has completely lost its load capacity. The structural

evaluation determines (1) the ability of the reactor coolant loop (RCL) piping to transfer (or redistribute) the RPV support loads to steam generator (SG) supports, reactor coolant pump (RCP) supports, and, if applicable, the concrete shield wall; and (2) the ability of SG and RCP supports to carry the additional loads transferred by the RCL piping.

The effects evaluation is conducted if the structural evaluation shows that the RPV support loads can be redistributed from the failed supports and that the SG and RCP supports are capable of carrying the additional loads. The effects evaluation then assesses consequences of the RPV motions such as, but not limited to, the ability to insert control rods for achieving hot shutdown and the ability of the reactor coolant pumps and any instrument lines and small diameter piping attached to the RPV to maintain their integrity.

2 STRUCTURAL EVALUATION

The objectives of the structural evaluation are to determine (1) the ability of the RCL piping to transfer the RPV support load to SG and RCP supports, and (2) the ability of SG and RCP supports to carry the additional loads transferred by the RCL piping. The structural evaluation assumes that the RPV has lost all four supports and that the RCL piping is not in contact with the concrete biological shield wall.

2.1 Load-transferring ability of RCL piping

The determination of the load-transferring ability of the RCL piping is based on a linear analysis following rules provided by Subsection NB and Appendix F, Division 1, Section III of the ASME Boiler and Pressure Vessel Code (ASME 1986). The same analysis also produces loads on SG and RCP supports, which can be used to evaluate the load carrying capability of the SG and RCP supports. The evaluation was conducted in two steps, namely, a preliminary evaluation and a final evaluation.

The preliminary evaluation was conducted based on an existing computer model of the RCL system of Unit 1 of the Zion nuclear power plant (Zion-1), which closely resembles the Trojan RCL system.

Two load combinations were evaluated: (1) the combination of dead weight, operating pressure, and the safe-shutdown earthquake, and (2) the combination of dead weight, operating pressure, and a loss-of-coolant accident. Both load combinations are classified as Level D Service Limits in accordance with the ASME Code. Thermal loads are not considered because thermally induced stresses are classified as secondary stresses by the ASME code and are not required to be considered by Appendix F evaluation. Static and dynamic linear analyses were conducted to comply with comparable rules specified by Subsection NB in conjunction with Appendix F, Division 1, Section III of the ASME Code. Results of this evaluation indicate that ASME Code Appendix F requirements are satisfied by each of the load combinations considered in the analysis, leading to the preliminary conclusion that the Trojan RCL piping is capable of transferring the RPV support loads to the SG and RCP supports. This evaluation is described in detail by Lu 1990.

Although the Trojan RCL system is very similar to the Zion- 1, there are differences as noted in Lu 1990. In order to confirm the results and conclusion of the preliminary structural evaluation, a Trojan RCL model was developed and, accordingly, a final structural evaluation was conducted. The final structural evaluation, which follows the same approach as the preliminary evaluation, generated results very comparable to those of the preliminary evaluation and, therefore, confirms the conclusion that the Trojan RCL piping is capable of transferring loads to the SG and RCP supports in the case of failure of the RPV supports.

It is noted that support stiffnesses for the Zion SG and RCP are used in the Trojan RCL model because we were not able to obtain the correct information. In order to validate our conclusion, a sensitivity study was conducted and subsequently concluded that the structural response of the RCL model is affected very little by the change in the SG and RCP supports.

2.2 Load-carrying capability of the SG and RCP supports

A structural analysis was conducted to determine the load-carrying capabilities of the lower SG supports and the RCP supports because such information is not available to us. The purpose of the structural analysis is to obtain lower bound estimates in accordance with the design basis loads the SG and RCP supports were originally designed for. The original design basis is a load combination which includes the dead load, the normal operation pressure and temperature, and a LOCA resulting from the rupture of one of the legs of the RCL piping. It is concluded from the structural analysis that the SG and RCP supports should have sufficient load margins because the lower bound load capacities exceed the loads transferred to the SG and RCP supports in case of the RPV support failure.

3 EFFECTS EVALUATION

In the case of a postulated RPV support failure, although loads will be transferred by the RCL piping and finally carried by the SG and RCP supports as demonstrated by the structural evaluation, the RPV will undergo movements considerably exceeding those originally restricted by the unfailed RPV supports. The movements consists of primarily a vertical translation (or drop) and a tilt from the vertical axis (resulting from the failure of three or less of the RPV supports). If the movements become excessive, they can lead to consequences that include initiating events that may not be mitigated by the engineered safeguard systems (ESS) and damages to the engineered safeguard systems. The effects evaluation identifies the potential initiating events and examines the engineered safeguard systems. An initial evaluation points out problem areas that were subsequently assessed by the final evaluation.

The initial effects evaluation, based on methodologies commonly employed by systems analysis, constructs several likely accident sequences that would occur as the result of the RPV movements. From the accident sequences, the potential initiating events and required safety systems were identified.

The evaluation identifies two initiating events of safety concern:

- (1) The multiple rupture of instrumentation thimble tubes or the guide tubes that penetrate the bottom head of the RPV could result in a LOCA that may lead to core uncover.

There are a total of 58 penetrations for high pressure conduits at the bottom head the RPV. These conduits are made of stainless steel with wall thickness of 0.25 in.—50 of them have a 0.4-in. inside diameter (ID) and 8 have a 0.6-in. ID. The conduits are also called thimble guide tubes because stainless steel thimble tubes (with wall thickness of approximately 0.1 in.) are placed inside these guide tubes. Carbon steel drive cables are inserted inside the thimble tubes.

During the normal operation, the annular space formed between the outer guide tube and the inner thimble tube is filled with the reactor coolant. Therefore, both the thimbles and the guide tubes are essentially the extensions of the reactor vessel pressure boundary. A hypothetical RPV support failure results in a downward movement of the vessel, which is normally restricted by the RPV supports. The downward movement could cause the rupture of multiple thimble tubes and the guide tubes, leading to core uncover due to a severe loss of core coolant.

- (2) The tilting of the RCP may affect the coastdown ability of the pump and the deformation of the RCP casing may cause the pump impellers to bind, resulting in loss of natural circulation.

The downward movement of the RPV can affect the RCP in two ways: (1) to cause tilting of the pump and (2) to cause deformation of the pump casing.

Tilting of the pump from the vertical axis can cause excessive vibration and in turn cause shutdown of the pump. In this situation, a reactor trip is required to avert core damage. It is important to reactor operation that the reactor coolant continues to flow (or coastdown) for a short time (approximately one minute) after reactor trip. In order to provide this flow with RCP power being shutdown, each RCP is equipped with a

flywheel. Thus, the rotating inertia of the pump, motor and flywheel is employed during the coastdown period to continue the reactor coolant flow.

Excessive deformation of the pump casing could bind the pump impellers, resulting in loss of natural circulation.

Other initiating events including LOCAs of various magnitudes, such as a LOCA caused by the rupture of the vent line at the RPV top head, and certain types of transients are possible but they are to be mitigated by the engineered safeguard systems (ESS's). The required ESS's are the reactor trip system and the emergency core cooling system (ECCS).

Evaluation of the engineered safeguard systems indicates that the instrumentation that generates the reactor trip signals is excore instrumentation and is not affected by the RPV movement and that the ability of generating signals to actuate the ECCS also will not be affected by the RPV movements. The evaluation, however, identifies the following safety concerns:

- (1) The possibility exists that the control rods could bind in the event of tilting of the RPV. The ability to insert control rods during a reactor trip would be adversely affected in this case.

The failure of all four RPV supports is the worst case in terms of loads to be transferred to the RCL piping and the SG and RCP supports, but the failure of three or less of the RPV supports can result in tilting of the RPV that may affect the ability to insert the control rods. The tilting of the RPV is considered to be limited by the concrete shield wall penetrations. The spacing between the RCL pipe and the penetration is approximately 6 in. and the distance from the penetration to the middle plane that contains the vertical axis of the RPV is about 175 in., resulting in a tilting angle of the RPV from the vertical axis approximately 2 degrees.

- (2) Two or more simultaneous ruptures of the safety injection lines could impair the ECCS function and lead to core damage.

Loss-of-coolant accidents other than the rupture of the main RCL piping and the rupture of the RPV are assumed to be mitigated by the ECCS. According to the success criteria for accumulator discharge, it is determined that at least two simultaneous rupture of the 10-in. safety injection lines would have to occur for core damage to occur.

Finally, a series of investigations were conducted to address the safety concerns identified by the initial effect evaluation as described below:

- (1) A nonlinear structural analysis was conducted to investigate the rupture of thimble guide tubes with 0.25 in. wall thickness and 0.6 in. inside diameter. These tubes are chosen for the analysis because they are the stiffer than the 0.4 in. ID tubes and, therefore, most susceptible to rupture due to the RPV movement.
- (2) An investigation was conducted to address the concerns with regard to the tilting of the RCP that may affect the coastdown ability of the pump and the deformation of the pump casing that may cause the impellers to bind.
- (3) An evaluation was conducted by Westinghouse Electric Corporation to determine the ability to insert control rods for the Trojan nuclear power plant with a tilted reactor vessel system.
- (4) A structural analysis was conducted to deal with the concern that the 10-in. safety injection lines may rupture as a result of the deformation of the RCL piping. The integrity of safety injection lines is required for the effective operation of the ECCS.

4 CONCLUSION

The Trojan nuclear power plant has been evaluated for the hypothetical failure of reactor pressure vessel supports due to radiation embrittlement. Several safety concerns have been identified by the evaluation. Further investigation, however, concludes that the failure of the Trojan RPV supports will not result in consequences of significant safety concern because:

- (1) The results of a structural evaluation indicate that the RCL piping is capable of transferring RPV support loads to the SG and RCP supports and that the SG and RCP supports have sufficient design margins to carry the additional loads transferred through the RCL piping.

- (2) A structural analysis of a typical thimble guide tube indicates that tube rupture will not be caused by the RPV movement.
- (3) Assessments of the reactor coolant pump indicate that tilting of the pump and the deformation of the casing would not lead to loss of its function during either the coastdown stage or the natural circulation stage.
- (4) Based on information provided by the NSSS vendor, the ability to insert control rods will not be affected as the RPV is tilted because of the RPV support failure.
- (5) An analysis of the safety injection lines demonstrates that the RPV movements will not cause rupture of these lines to cripple the ECCS function.

It is noted that the scope of this evaluation is limited to consequences that can uniquely occur as the results of a hypothetical failure of the RPV supports. This study does not consider other consequences such as random failure of equipment or equipment unavailability due to inappropriate maintenance or testing, or other complications caused by human error factors, which are normally considered in a more complete probabilistic risk assessment of nuclear power plants.

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