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ANALYSIS OF LOSS OF DECAY-HEAT-REMOVAL SEQUENCES AT BROWNS FERRY UNIT ONE*

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This paper summarizes the Oak Ridge National Laboratory (ORNL) report† "Loss of DHR Sequences at Browns Ferry Unit One - Accident Sequence Analysis" (NUREG/CR-2973) The Loss of DHR investigation is the third in a series of accident studies concerning the BWR 4 - MK I containment plant design. These studies, sponsored by the Nuclear Regulatory Commission Severe Accident Sequence Analysis (SASA) program, have been conducted at ORNL with the full cooperation of the Tennessee Valley Authority (TVA), using Unit One of the Browns Ferry Nuclear Plant as the model design. Each unit of this three-unit plant has a maximum authorized power of 3293 MW(t) or 1067 net MW(e). The primary containments are of the Mark I pressure suppression pool type and the three units share a secondary containment of the controlled leakage, elevated release design. Each unit occupies a separate reactor building located in one structure underneath the common refueling floor.

The purpose of the SASA studies is to predetermine the probable course of postulated severe accidents so as to establish the timing and the sequence of events. The SASA studies also produce recommendations concerning the implementation of better system design and better emergency operating instructions and operator training. In the interest of efficiency, it is desirable that the SASA effort be directed toward the dominant accident sequences identified by probabilistic risk assessment (PRA) techniques. The ORNL studies also include a detailed, best-estimate calculation of the release and transport of radioactive fission products following postulated severe accidents.

The Loss of Decay Heat Removal (DHR) accident sequences were selected for study by the SASA program because they constitute six of the eight dominant accident sequences leading to core melt which have been identified for Browns Ferry Unit One by the NRC's Interim Reliability Evaluation Program (IREP). The IREP study is a PRA whose function is to attempt to consider all possible accident sequences at a nuclear plant using event tree and fault tree methodology for the purpose of identifying the more probable, or dominant, sequences. The SASA approach, on the other hand, is to examine a particular category of accident sequences in far greater depth than would be possible in a PRA study.

The basic initiating events for a Loss of DHR sequence include a reactor scram, closure of the main steam isolation valves (MSIVs) so that the main condenser cannot function as a heat sink, and subsequently, failure of the RHR system to provide either suppression pool cooling or reactor vessel shutdown cooling. The steam produced by decay heat is relieved from the reactor vessel by the safety/relief valves (SRVs) and is condensed in the pressure suppression pool. The suppression pool temperature increases monotonically and the resulting increase of pressure in the primary containment can ultimately cause failure of the primary containment, with the attendant possibility of severe fuel damage.

Multiple injection systems would be available after a loss of DHR accident. Reactor vessel water level can be maintained during the early stages of the accident by operation of either the high pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC) pumps. After 4 h, the 10.7 ℓ/s (170 gpm) injection provided by the control rod drive (CRD) hydraulic system pump is sufficient to maintain the vessel water level. All three pumps take suction on the condensate storage tank, and operating procedures provide that there would be an initial supply of water in the tank sufficient to last well beyond the time of containment failure in a Loss of DHR accident sequence.

The BWR-LACP code developed at ORNL for BWR analysis has been used for the analysis of the sequence of events before containment failure. This code employs efficient coding to assess the effect of operator actions or system equipment failures on the thermal hydraulic conditions within the reactor vessel and containment. The assumption of uniform suppression pool temperature is built into the BWR-LACP calculation. This assumption is approximately true providing that at least one pump and basic piping loop of the RHR system is available for circulation and mixing of the suppression pool water.

For the case of a Loss of DHR accident sequence with RHR pump operation and uniform heatup of the pressure suppression pool, the containment pressure exceeds the static overpressurization failure pressure of 0.910 MPa (117 psig) after 35 h. This failure pressure was predicted by a detailed study of static overpressurization failure of the Browns Ferry steel containment (Ref. L. D. Greimann et al., "Reliability Analysis of Steel Containment Strength," NUREG/CR-2442, Iowa St. University Ames Laboratory, June 1982). The study predicted that the failure would occur at the interface between the drywell spherical and cylindrical sections.

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†Investigation of fission product release and transport following Loss of DHR-initiated severe accident sequences is underway; release of a companion report is planned for the future.

For the case without RHR pump operation and therefore with non-uniform heatup of the suppression pool, a special computer code was developed to predict the pool temperature distribution. The results indicate that the pool becomes thermally stratified, with hotter water in the top layers, and that the hottest water exists in the bay of a discharging SRV quenching device. The net effect of the temperature non-uniformities is a slightly faster build up of primary containment pressure, in which the assumed static overpressure failure point is reached after 28 h, instead of after 35 h.

A variety of possible scenarios can be initiated by the drywell failure. A large rupture would rapidly release about 35 h of stored decay heat energy into the reactor building, and this might cause the failure of all vessel water injection. In order to evaluate the consequences of such a scenario, MARCH computations were initiated just before the drywell reached the failure pressure, with initial conditions provided by the results of the BWR-LACP code at the 34 h point. All water injection to the reactor vessel is assumed to cease after containment failure but the primary coolant system is assumed to maintain its integrity during and after the primary containment blowdown. A pressurized boiloff of the water in the reactor vessel at the time of containment failure follows. Because of the large inventory of water in the reactor vessel that must be boiled away through the relief valves, and the low level of decay heat this long after shutdown, core uncovery does not start until about 2 h after the loss of injection. The onset of fuel melting occurs about 3 h later, or 39.5 h after the inception of the accident.

The integrity of the reactor building and refueling bay is maintained by installed systems. The building blowout panels prevent excessive internal pressure by relieving the outflow of steam that accompanys drywell failure. The reactor building fire sprays would be actuated by the high atmosphere temperatures caused by the drywell blowdown. After the blowdown of the drywell, the Standby Gas Treatment system maintains a net flow of outdoor air into the reactor building. For the postulated loss of all vessel water injection sequence, core-concrete interaction products would reach a concentration sufficient for initiation of a burn in the reactor building between 9.6 and 10.7 h after drywell failure. The effects of this possible CO/H_2 burn would be mitigated by the building fire sprays.

The results of this study illustrate the characteristic slow nature of the Loss of DHR accident sequence and the very long time available for the operator to take corrective action. Recovery of the pool cooling mode of any one of the four RHR heat exchangers would arrest the increase of pool temperature and thus prevent primary containment failure. If pool temperature had advanced to over 100 degrees C (212 degrees F) at the time of recovery of pool cooling, operator action to throttle the output of the RHR pumps would provide assurance of an adequate net postive suction head for the pumping of the hot suppression pool water.

Other less conventional recovery strategies were investigated, and found to be feasible. For example, the addition of river water via primary containment sprays would prevent containment failure in this sequence. It is uncertain that this type of mitigation would be applied during an actual accident sequence because the requisite operating procedures have not been developed.

One of the major findings of the study concerns the role of the CRD hydraulic system. During the Loss of DHR accident sequence the CRD hydraulic system functions as a small capacity injection system. The decay heat level decreases after the initiating reactor scram and, after 4 h, the CRD hydraulic system is capable of supplying all required cooling water to the reactor vessel. Even if an operator caused premature failure of the RCIC injection system by shifting suction to the hot suppression pool, the CRD system, which takes suction only on the cool water in the condensate storage tank, would maintain sufficient injection to the reactor vessel. This fact is usually neglected in PRAs. Due to its potential importance to plant safety, the CRD hydraulic system should be included in risk analyses and should be a part of plant operator training.

The sequence of events determined in this study by best-estimate calculations is considerably different from the sequence presented in the IREP report. The IREP investigators concluded that a loss of decay heat removal would lead to a loss of vessel water injection, followed by core melt within about 8 h and that the containment failure would occur after the core melt. ORNL investigators concluded that, for the same sequence, failure of vessel water injection prior to containment failure would be very unlikely, and that vessel water injection might continue to be available even after containment failure. This disagreement is due not to the use of different calculational models, but to different assumptions regarding the quantity of stored cooling water and the capabilities of the installed pumping systems. If a more refined PRA, taking into account the ORNL results, were undertaken, it is believed that the results would show that the Loss of DHR sequences do not dominate the overall risk.

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