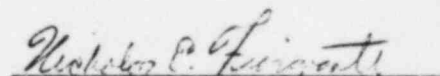


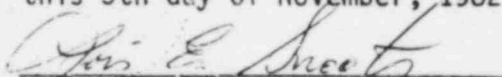


4. As described in NUREG-0785 the postulated pipe break in the scram discharge volume if not isolated would be the equivalent of a small unisolated break in the bottom of the reactor vessel. The coolant inventory flowing from that break would be lost from the primary containment and thus would not accumulate in the drywell-torus that is the normal reservoir for water for long-term cooling. Moreover, in that NUREG-0785 scenario the lost coolant could possibly be directed to areas of the reactor building housing the emergency core cooling equipment.
5. The scenario described in NUREG-0785 is not applicable to the BWR6/Mark III containment design and thus an SDV pipe break in nuclear plants of that type poses no threat to the long-term cooling capability provided by the ECCS. See NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," at p. 2-3.
6. In the BWR6/Mark III design, the scram discharge volume piping is located within the primary containment. Any primary coolant inventory released from a scram discharge volume pipe break would remain in containment and be returned directly to the suppression pool. Any of the released primary coolant inventory that flashes to steam would be condensed inside containment and also returned to the suppression pool. This primary coolant inventory will not be lost from the primary containment and thus can be re-used for makeup to the vessel. Using the suppression pool as a water source, the ECCS has the capability of providing makeup water to the vessel until the break can be isolated.

7. Immediate isolation of the break in the scram discharge volume piping is not necessary because the ECCS, the residual heat removal (RHR) system and the reactor core isolation cooling (RCIC) system can prevent the core from being uncovered and can remove the decay heat from the core for an extended period of time. Moreover, if the scram discharge volume pipe break cannot be isolated remotely by closing the scram outlet valves from the control room, manual isolation of the break can be accomplished when the dose rate decays sufficiently to allow entry into containment.
8. The ECCS, RHR and RCIC are located in individual watertight compartments outside of the primary containment in the auxiliary building. The ECCS, RCIC and RHR would not be subjected to flooding or any adverse environmental conditions resulting from a postulated pipe break in the scram discharge volume piping. The ECCS, RCIC and RHR have the capability of being automatically or manually actuated to mitigate the consequences of a pipe break in scram discharge volume piping.
9. Therefore, a SDV pipe break in the Perry plant will not cause an unrecoverable loss-of-coolant accident that threatens the safety of the reactor.

  
Nicholas E. Fioravante

Subscribed and sworn to before me  
this 5th day of November, 1982

  
Notary Public

My Commission Expires: 7-1-1986

NICHOLAS E. FIORAVANTE  
PROFESSIONAL QUALIFICATIONS

AUXILIARY SYSTEMS BRANCH  
DIVISION OF SYSTEMS INTEGRATION  
OFFICE OF NUCLEAR REACTOR REGULATION

I am a Mechanical Engineer in the Auxiliary Systems Branch in the Division of Systems Integration, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. In this position, I perform technical reviews and evaluations of the functional capability of auxiliary systems and components pursuant to the construction and operation of reactors.

I received a Bachelor of Science Degree in Mechanical Engineering from George Washington University in 1977. Since 1977, I have taken courses on PWR and BWR technology, System Reliability Engineering and Risk Assessment, Effects of Human Performance on Nuclear Power Plant Operations, Accident Phenomenology and Containment Response and Bayesian Reliability Analysis. I'm presently enrolled in Master of Science in Management degree program with Frostburg State College.

My experience includes three years with the David W. Taylor Naval Ship R&D Center as a Structural Engineer engaged in various design phases of naval ship protection systems.

I joined the Auxiliary Systems Branch in May, 1980. Since that time, I have prepared safety evaluation inputs for the Perry Nuclear Power Plant; the Clinton Power Station, the Shoreham safety shutdown review, the Indian Point Stations' fuel pool expansion program, the Big Rock Point loss of service water review and the auxiliary feedwater

reliability reviews of D.C. Cook, Beaver Valley 1, and San Onofre 1. I prepared revisions of Standard Review Plan sections 9.3.1, 9.3.3, 9.3.5 and 3.6.1. Additionally, my assignments include technical monitor and project manager responsibility for the "Post-Fire Shutdown Capability" program.

I have responsibility for the review of the following nuclear power plant auxiliary systems: new and spent fuel storage; spent fuel cooling system; spent fuel handling; service water system; reactor auxiliary cooling water system; demineralized water makeup system; ultimate heat sink; condensate storage facilities; compressed air system; standby liquid control system; HVAC system for control room area, spent fuel pool area, auxiliary and radwaste area, and ECCS areas; main steam supply system and circulating water system.



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