

failures which he argues could lead to a severe accident or escalate an accident to a more severe one. He also contends that the Commission is unable to predict the impact of such multiple failures.

The Commission's Rules and Regulations do not explicitly require that plants be designed to accommodate multiple failures. However, inherent in the required analyses are conservative assumptions that effectively provide assurance that the plant can safely accommodate a spectrum of multiple failures scenarios. For example, GDC 26 requires that transients and accidents be analyzed assuming the most reactive control rod stuck in the withdrawn position. Other transients and accidents assume the loss of offsite power. These assumptions, when combined with the single failure assumption, essentially provide for multiple failure scenarios being evaluated as design bases. In addition to these assumptions, the staff also believes, as a result of many diverse studies (such as USIs, GIs*, etc.) that many multiple failure scenarios can be accommodated by present plant designs. For example, we have high confidence that plants such as Catawba can withstand multiple steam generator tube ruptures, despite the fact that the design basis considers only one tube rupturing. Protection against multiple failures is also provided by the emergency operator guidelines. These guidelines specifically instruct the operator on how to manage plant transients and accidents that could involve multiple failures.

Finally, the staff has reviewed and/or performed a number of plant specific probabilistic risk assessments (PRAs) for a spectrum of plant types. None of these PRAs has uncovered any multiple failure scenarios

*Unresolved Safety Issues and Generic Issues.

of sufficiently high probability or risk to warrant any design changes in any plant, either operating or under license review.

All of Dr. Kaku's concerns have been adequately considered by the Staff. The following is a brief description of the licensing process that addresses the above concerns.

4. There are several layers of protection inherent in the design and operation of a nuclear power plant such as Catawba.

A. First, in the design stage, the safety analyses are performed for a wide spectrum of transients and accidents, i.e., overcooling events and overheating events, overpressurization at low temperature and at high temperature operation, increased and decreased reactor coolant inventory, partial and total loss of reactor coolant flow, and reactivity and radiological release accidents. These safety analyses utilize a set of assumptions and initial conditions that will produce conservative estimates of the severity of the consequences of the analyzed accident. Some examples of these assumptions are (a) the use of 102% initial power level, (b) the use of moderator temperature-reactivity coefficients that are conservative with respect to the entire life of the plant, (c) taking credit only for the "safety grade" equipment to mitigate the accidents, and (d) assuming that the worst single failure occurs in any component or system expected to actuate during the course of the event. For each transient or accident type a spectrum of events is evaluated (e.g., reactor coolant system breaks that range from a minor leak just in excess of the normal makeup system capacity to a double ended guillotine break that will depressurize the reactor coolant system and uncover the core.)

Each type of event is bounded by analyses that preclude the need to analyze a multitude of scenarios that are somewhat different from one another. The computer models used by the licensee for accident analyses have undergone Staff review and have been approved for their intended use as discussed in (a) through (d) above. Each of these codes is composed of three essential parts (a) the methodology and modeling, (b) its application and limitations, and (c) the code verification. Parts (a) and (b) determine the applicability and limitations of each code. Part (c) provides verification of the adequacy of the code by a variety of means that include some or all of the following (i) benchmarking against other codes, (ii) comparison with plant operational experience, and (iii) prediction of test results, e.g., the semiscale and LOFT experiments.

B. Second, for plant operation the licensee relies on several documents that include (1) the plant Technical Specifications, (2) the normal operating procedures, and (3) the plant emergency operating procedures. The licensee is committed to adhere to the requirements of each of these documents. The first two documents above set upper and lower limits on the plant conditions and the rate of change of those conditions during normal operation such that thermal or mechanical stresses as well as undesirable nuclear behaviors will be minimized. The third document, the emergency operating procedures, came under a great degree of scrutiny by the Staff, especially after the TMI-2 accident (see NUREG-0737, I.C.1). The Staff does not routinely review plant specific emergency procedures. However, the staff does review the emergency procedures guidelines. The analyses performed in support of development of the emergency procedures guidelines account for operator errors and multiple failures. For operator errors, these procedures have several cross checks of system parameters

that the operator is directed to conduct so that an operator error of omission or commission will be identified and corrected. As an example, if an operator commits an error by terminating the high pressure safety injection (HPSI) without observing the HPSI-termination criterion, the operator is directed to check the backup HPSI-reinstatement criterion which provides instructions to again turn on the HPSI. This logic is adopted throughout the emergency procedures guidelines. As for multiple failures, many of the supporting analyses account for multiple failures. Examples of these include a combined steam generator tube rupture and a steam line break accident, a combination of loss of main feedwater and auxiliary feedwater systems, and a total loss of HPSI with a total loss of AC power.

5. Dr. Kaku expresses concern in paragraph 8 of his affidavit about primary system coolant contamination by small microleaks in the fuel rods in conjunction with a steam generator tube rupture event. These events are addressed in the licensee's FSAR and the Staff's safety evaluation report (SER) and its supplements. The Catawba plant, as well as any other nuclear power plant, is required to maintain the primary system iodine concentration below the Technical Specification value, or else be shutdown. The plant safety analysis was conducted assuming the iodine concentration was at the Technical Specification limit. Thus, the effects of any fuel rod leakage have been adequately accounted for in the analyses.

6. Dr. Kaku expresses concern in paragraph 9 of his affidavit about fuel handling accidents. This is an accident that has been analyzed in the FSAR and reviewed by the Staff as shown in the SER. This type of accident, however, is only credible in a refueling outage when irradiated fuel is manipulated.

7. Dr. Kaku states in paragraph 10 of his affidavit that a scrammed reactor is left with a decay heat equivalent to 5% of full power, which is claimed to be sufficient to melt the fuel. However, his assertion greatly exaggerates the amount of decay heat available, which is on the order of 3% after a few minutes and much less thereafter. In addition, the accident scenarios referenced in paragraph 10 which lead to fuel heatup are highly unlikely considering the low decay heat available, the amount of primary and secondary water present and the time available to the operator to assess the situation and take the appropriate corrective actions. In addition, there are safety grade makeup systems for the primary and secondary systems. In summary, at decay heat levels, no fuel melting can occur as long as the fuel rods remain covered with primary coolant.

8. In his paragraph 15, Dr. Kaku challenges the probability calculations in WASH-1400 because of the asserted neglect of common and multiple mode failures. However, the WASH-1400 study did indeed consider common and multiple mode failure effects where appropriate in deriving probabilities. Notwithstanding, WASH-1400 was not used by the Staff as a licensing basis in the review of Catawba plant.

9. Dr. Kaku also expresses concern about cascading pipe failures. This is presumably caused by the pipe whip, although Dr. Kaku does not suggest a mechanism. This problem is considered in all piping designs and carefully reviewed by the NRC. Multiple piping breaks under any circumstance are considered extremely unlikely.

10. Dr. Kaku discusses the possibility of a Class VIII large-break loss-of-coolant accident (LOCA) "sliding" into a Class IX event because of HPSI system failure. First, it should be noted that mitigation of a

large-break LOCA does not rely on HPSI. In any event, all emergency core cooling systems (ECCS) (HPSI, low pressure safety injection and passive accumulators) are redundant. For all required design basis ECCS analyses, the failure of at least one of each required ECCS system is assumed. The probability of failing all trains of the required ECCS, whether by an earthquake or otherwise, is extremely small.

11. In his paragraph 16, Dr. Kaku is concerned about multiple failures and provides an example, a large-break LOCA with total failure of the HPSI system. However, failure of the HPSI system is not sufficient to cause fuel melting in the case of a large-break LOCA. The low pressure sources of water and containment heat removal are sufficient for accident mitigation.

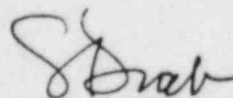
12. Finally, Dr. Kaku is concerned about the integrity of the reactor containment and in particular the ice-condenser type containment and the likelihood of its failure due to overpressurization or explosions inside the containment. The Staff has extensively reviewed missile generation by a steam explosion since publication of WASH-1400. While steam explosions both in-vessel and ex-vessel are postulated to occur when molten core materials contact water, the energy available from these explosions is not sufficient to generate debris of such a size and energy as to cause failure of the containment. Furthermore, the Staff has fully reviewed the design of the Catawba ice-condenser containment and found it to be adequate. The ice-condenser containments' walls are relatively thinner than the conventional containment types because of the permanent heat sink (the ice baskets) that absorbs large amounts of the heat released after a pipe-break inside containment. As a result, the ice-condenser containment walls are never exposed to the same post accident environs as their conventional containment counterparts.

13. In addition to relying on passive heat absorbers to suppress pressure, the ice-condenser design relies on hydrogen igniters to reduce the likelihood of large hydrogen burns. Contrary to the implication of Dr. Kaku, the study he cites (NUREG/CR-1059, Vol. 1, p. 4-8) found that overall public risk resulting from a typical PWR with ice condenser containment is expected to be similar to that of a dry containment.

14. Finally, Dr. Kaku asserts that the ice condenser feature at Catawba has never been tested under accident or simulated conditions. A set of full scale ice-condenser basket tests was conducted by Westinghouse (Waltz Mill tests) in the late 1960s and early 1970s. The Staff has found the tests results acceptable (see NUREG-0474, July 1978).

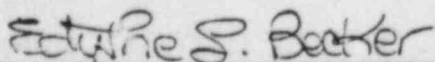
15. In sum, Dr. Kaku raises numerous questions of a general nature challenging the Staff's analysis of the safety of plant reactor systems. However, these questions have been accounted for in Staff safety evaluations, and determined not to present any safety concern.

16. Finally, Dr. Kaku does not point to any safety-related component or structure at Catawba which is unsafe, or whose design and function was not adequately evaluated by the Staff. I conclude that Dr. Kaku does not demonstrate that any Catawba reactor systems component will fail to operate so as to pose a significant risk to the health and safety of the public.



Sammy S. Diab

Subscribed and sworn to before me
this 3rd day of January, 1985.



Notary Public

My Commission expires:

2/1/86

STATEMENT OF PROFESSIONAL QUALIFICATIONS

SAMMY S. DIAB

I am a Nuclear Engineer in the Reactor Systems Branch of the U.S. Nuclear Regulatory Commission (NRC). In this position, I am responsible for the technical analysis and evaluation of reactor systems, accidents and transients, and applications for nuclear reactor operating licenses. I have been in my current position since 1980.

From 1978 to 1980, I was a reactor systems reviewer in the Reactor Safety Branch, Division of Operating Reactors of the NRC. In that position my responsibilities included: systems analyses, accident and transient analyses, and reload application reviews.

From 1977 to 1978, I was a Nuclear Engineer in the Engineering Methodology Standards Branch, Office of Standards of the NRC. In that position I was responsible for updating and revising the standard review plan. I developed Regulatory Guide 1.139, "Residual Heat Removal Guidance", and Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems".

From 1973 to 1977, I was a Nuclear Engineer with Bechtel Power Corporation, Gaithersburg Power Division, Maryland. I was responsible for reactor containment pressure and temperature analyses following a spectrum of high energy line breaks, jet impingement calculations, and subcompartment transient behavior. I developed and used computer codes. I also modified existing computer codes.

From 1971 to 1973, I was a research assistant with the Nuclear Engineering Department of the Pennsylvania State University. In 1974 I was awarded a M.S. degree in Nuclear Engineering from Pennsylvania State University.

From 1967 to 1971, I was a researcher with the Egyptian Atomic Energy Establishment. I received my B.S. degree in Nuclear Engineering from the University of Alexandria, Egypt.