UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-454 - OL
	)	50-455 - OL
(Byron Station,	)	
Unit Nos. 1 and 2)	)	

AFFIDAVIT OF WALTON L. JENSEN, JR.
REGARDING DAARE/SAFE CONTENTION 4

I, Walton L. Jensen, Jr., being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Senior Nuclear Engineer in the Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached.
2. I am responsible for the technical analysis and evaluation of the public health and safety aspects of reactor systems. The purpose of this affidavit is to address the Staff position on DAARE/SAFE contention 4 dealing with multiple independent failure accidents.
3. Contention 4 alleges, in general, that multiple independent failure accidents must be considered in nuclear power plant design and provides 15 specific hypothetical examples of such events. In his March 12, 1982 deposition, Dr. Michio Kaku, the Intervenor's witness on this issue, added three more examples to the list (16-18). The contention defines such accidents as those "which occurred in proxi-

mate time to one another without actually being caused by one another." The Staff has accepted that definition for purposes of arriving at a position on this contention.

4. The Commission's regulations do not require explicit design consideration of multiple independent failures per se. They do require consideration of multiple failures resulting from a single occurrence for systems covered by certain specific GDC (e.g., GDC 21, 34, 35, 38, 44). Specifically, these requirements provide that, in addition to an initial single failure or accident event, the plant be designed to withstand single failures in systems required to mitigate the event. Several of the hypothesized events in contention 4 constitute multiple independent failures which exceed the single failure requirement incorporated into the applicable General Design Criteria. Examples 2-7, 8-9, 16-17 fall into this category. Examples 2-7 further constitute ATWS events which are the subject of Commission rulemaking and for which interim requirements have been developed and imposed upon the Applicant. Hypothesized events 1 and 11-12 appear to be multiple dependent failures which are discussed in Attachment A and require no FSAR analysis as it has been determined that these will not occur. Examples 10, 13 and 18 constitute single failures which have already been evaluated in the FSAR and SER. The Staff has concluded that these events will not result in consequences which exceed regulatory limits. Example 14 cannot be properly classified as a multiple independent failure of equipment since it entails improper operator action. Upgraded operator requirements have been applied and favorably reviewed by the Staff for this

facility. Example 15 is vague. Unspecified instrument or gauge failures are postulated to result in some unspecified accident. However, as discussed in Attachment A, failure of multiple redundant and supportive instrumentation would be required to create such an accident situation. Hence, such an event is beyond the single failure requirement to consider and is furthermore unlikely. The Staff position on all the examples is given in Attachment A.

5. The Commission's principal design criteria for nuclear reactors are contained in 10 C.F.R. Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." These criteria establish the "necessary design, fabrication, construction, testing and performance requirements for structures, systems and components important to safety; that is structures systems and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." _Ibid._, Introduction.

6. A fundamental element of the GDC is the single failure requirement. The single failure requirement is defined in Appendix A as follows:

   > an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure.

7. The implementation of this requirement resulted in the development of a set of Design Basis Events (DBE) described in the Standard Review Plan (SRP) (NUREG-0800). The designation of DBEs was a matter of the collective professional judgment of the Staff, industry and non-industry experts, and the ACRS. An effort was made to bound those events that might reasonably be expected to happen. This was

accomplished principally by postulating single failures in each major plant system, in turn, and requiring that the plant be capable of managing the consequences without undue risk to the public health and safety. The transient and accidents analyzed are representative of classes of events that have been judged to be of significant severity and sufficient likelihood to require consideration. Similarly, the associated analysis methods and acceptance criteria are also not realistic, but are conservative, or bounding representations of actual or expected conditions.

8. A list of the design basis events required to be analyzed by the Standard Review Plan by operating license applicants and analyzed by the applicant for Byron in Chapter 15 of the FSAR, is provided in Attachment B. With the exception of items 10, 13, and 18 as noted in paragraph 4, the hypothetical events given in Contention 4 are not among the list of design basis events.

9. Additional failures are postulated to occur in systems relied upon to mitigate the event, as discussed in GDC 21, "Protection System Reliability and Testability," GDC 34, "Residual Heat Removal", GDC 35, "Emergency Core Cooling," GDC 38, "Containment Heat Removal" and GDC 44, "Cooling Water."

10. Structures, systems and components which are relied upon to mitigate design basis events are regarded by the NRC staff as safety related. They must be designed in accordance with the applicable GDC if they are required to fulfill the critical safety functions defined in 10 CFR 100, Appendix A. This consists of those structures, systems, or components necessary to assure: "(1) the integrity of the reactor

coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of this part." 10 C.F.R. Part 100, Appendix A, Section III. This latter definition includes the containment building and that equipment such as piping and valves used to isolate the containment from the outside environment to the extent necessary to maintain radiological release within the limits of 10 C.F.R. Part 100 following an accident.

11. Systems which are required to meet safety related criteria for Byron include (but are not limited to) the Reactor Protection System, which provides for the termination of the chain reaction fission process within the reactor core, the ECC systems, which provide for restoration of core cooling in the event of a LOCA, the safety valves, which protect the reactor system pressure boundary from overpressure and the auxiliary feedwater and residual heat removal systems, which provide for removal of decay heat in shutting down the reactor to a safe shutdown condition. These safety related systems are designed in accordance with the General Design Criteria, as implemented by the various specific Standard Review Plans of NUREG-0800, to:

(1) withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions;

(2) accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance,

testing, and postulated accidents, including loss-of-coolant accidents;

(3) be protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit; and

(4) include sufficient redundancy, independence, and separation to insure that required safety functions can be performed even with the most limiting single failure.

12. The conservative accident analyses discussed in Chapter 15 of the FSAR are used to demonstrate that the potential consequences to the health and safety of the public are within acceptable limits for a wide range of postulated events even though specific actual events might not follow the same assumptions made in the analyses. In addition, the analyses performed are used to demonstrate that the potential consequences to the health and safety of the public are within acceptable limits when only safety- related equipment and systems are used to mitigate the consequences of the postulated event.

13. As was discussed in paragraph 7, the plant design has been reviewed for a spectrum of transients and accidents that are representative of classes of events that have been judged to be of significant severity and sufficient likelihood to require consideration. It is not required nor possible to analyze or even define all possible accident failure sequences for any nuclear power plant. But assuring an acceptable level of reactor safety is not limited to the analyses of a number of transients and accidents and ascertaining that the plant is designed to control and/or accommodate the consequences of these events. Adequate safety also depends on a "defense in depth"

approach which recognizes the availability of a large number of plant design features as well as the availability of well trained operators using carefully prepared procedures. The cornerstone of the "defense in depth" philosophy is the use of multiple, successive barriers to the escape of radioactivity and assuring that these barriers are not compromised as the result of transients and accidents.

14. The first level of protection deals with design for safety in normal operation and tolerance for system malfunctions. It emphasizes quality, redundancy, and inspectability. Criteria and requirements applied to the structures, systems and components needed for normal operations (e.g., primary pressure boundary, main feedwater system, main steam system, turbine, radiation monitoring system, effluent control system, the control room and control systems) are found in regulatory guidance documents and in the Standard Review Plan, Regulatory Guides, General Design Criteria (10 CFR Part 50, App. A). The second level of protection assumes that incidents will occur in spite of care in design, construction and operation. It requires the provision of systems to detect incipient failure and to shutdown the plant so as to prevent or minimize damage when such incidents occur. The third level of protection assumes the occurrence of damaging accidents. It requires the provision of safety systems to limit or control the consequences of hypothetical accidents (e.g., loss of coolant accidents). In designing these safety systems, we require the assumption of a large fission product release per 10 CFR Part 100, where some protective systems are assumed to degrade or

fail simultaneously with the accident they are intended to control. The reactor fuel cladding, the reactor coolant system pressure boundary, and the reactor containment building constitute the key parts of the third level of defense in depth. The fuel clad and the reactor coolant system boundary are designed to contain the radioactive fission products produced in the core; likewise the reactor containment building and associated isolation equipment, which are safety-related, are designed to limit radioactivity releases and serve as the final boundary to the outside environment. The analyses in Chapter 15 of the FSAR, which are reviewed by the NRC staff, are used to verify that the integrity of the cladding, reactor coolant system boundary, or both, are maintained within specified limits following the postulated design basis events discussed earlier.

15. Another level of protection is provided by the trained operator and the emergency operating procedures. The operator, utilizing the procedures, is trained to take actions to maintain the plant in a safe condition independent of the type or number of equipment or system failures which might occur. In performing the key safety functions, the operator is instructed to use any and all equipment or systems which might be available whether it is safety-related (safety-grade) or not. In addition to the design basis events, analyses assuming various event sequences (including multiple failures) that could occur and fall outside of the required design envelope have been utilized in the preparation of the emergency operating procedures. This approach for the plant operators is a result of the lessons learned from the TMI-2 accident. Its

objective is to further assure that the operator is able to respond to the complete spectrum of possible events. It would be impossible to assume that an operator could memorize all multiple failure sequences, and rapidly diagnose the actual, specific event. Therefore, the approach we use is to guide the operator to recognition of certain symptoms of events, and to respond to symptoms rather than to a specific event. This involves "all" events being broken down into categories that are all inclusive, e.g., loss of heat sink, overcooling, loss of inventory, reactivity. Operators are trained and procedures are written to treat symptoms of these categories and gain control of the plant no matter what combination of failures caused the particular event. This approach is being implemented at the Byron plant.

16. In summary, the analyses in Chapter 15 of the FSAR combined with the "defense in depth" approach, which has been extended to include multiple failures outside of the required design basis in the emergency operating procedures, and compliance with approved regulatory guidance, constitute the methodology used to ensure that nuclear power plant operation will not result in undue risk to the health and safety of the public. It was never intended nor is it necessary to analyze all possible accident sequences to assure an adequate level of safety.

17. Contention 4 mischaracterizes key elements of the TMI accident in drawing its conclusions about the necessity to consider certain hypothetical accidents in plant design. The accident at TMI-2 was caused by a failure in the main feedwater system which caused the

opening of a pressurizer relief valve on high reactor system pressure which then failed to close. An incorrect operator action in prematurely terminating ECCS led to reactor core damage. In addition, though not contributing to the severity of the event, the auxiliary feedwater system failed to be immediately actuated. The AFW system was restored by the operator approximately nine minutes into the event. The auxiliary feedwater system was not classified as safety-related at TMI-2. AFW, however, is safety-related at Byron.

18. No designed safety-related system failed to perform its function at TMI-2 as the result of mechanical failure. The only failure of a designed safety-related system at TMI-2 was that of the ECCS which was the result of incorrect operator action. Had the operator permitted the ECCS to fulfill its function, the event would have been within the plant design basis and no core damage would have occurred.

19. Following the TMI-2 event, the NRC staff imposed additional requirements on nuclear licensees and applicants to:

    (1) develop improved procedures and training in dealing with transients and accidents;

    (2) improve instrumentation available to the operator in dealing with transients and accidents;

    (3) improve the reliability of the auxiliary feedwater system and

(4) reduce the likelihood of excessive coolant loss from stuck open primary system relief and safety valves.

These requirements are contained in NUREG-0737.[1/] The applicant has addressed these requirements in the Byron FSAR. The Staff has reviewed this matter as contained in the SER. See SER Table 1.1. The applicant will be required to implement all NUREG-0737 requirements applicable to Westinghouse designed PWRs.

20. As relevant to accident mitigation, the applicant is working with the Westingouse Owners' Group to develop Emergency Response Guidelines which will consider multiple failure events in response to Item I.C.1 of NUREG-0737. Among the multiple failure events to be included are:

(1) multiple tube ruptures in a single steam generator and tube rupture in more than one steam generator;

(2) failure of main and auxiliary feedwater;

(3) failure of high-pressure reactor coolant makeup system;

(4) an anticipated transient without scram (ATWS) event following a loss of offsite power, stuck-open relief valve or safety/relief valve, or loss of main feedwater; and

---

1/ Clarification of TMI Action Plan Requirements (November 1980).

(5) operator errors of omission or commission.

The Emergency Response Guidelines will be developed into plant specific procedures for Byron. Operators will be trained to use these procedures by startup.

21. The Staff believes that with the exception of examples 10, 13, and 18 which are already considered in the Byron FSAR, consideration of the majority of hypothetical accident examples in contention 4 for plant design purposes is beyond the single failure requirement, which does not require analysis of multiple independent failures. Examples 2 through 7 constitute ATWS events which are the subject of proposed rulemaking and for which interim requirements have been developed and imposed on the Applicant. Finally, contrary to Contention 4, the Byron FSAR is in compliance with 10 C.F.R. § 50.34(b)(4) regarding the consideration of timely and pertinent information relevant to the evaluation of design and performance of structures, systems and components provided for the prevention of accidents and the mitigation of consequences.

The foregoing and accompanying qualifications statement are true and correct to the best of my knowledge.

Walton L. Jensen, Jr.

Walton L. Jensen, Jr.

Subscribed and sworn to before me this 1st day of June, 1982.

Judy L. Butts Notary Public

JUDY L. BUTTS
NOTARY PUBLIC STATE OF MARYLAND
My Commission Expires July 1, 1982

ATTACHMENT A

The Staff position on the hypothetical accident examples raised in Contention 4 is provided below:

Example (1) Rupture of a defective control rod drive mechanism (CRDM) housing which causes adjacent, similarly defective CRDM housings to rupture in a cascade manner. Such ruptures could cause the effected control rods to be ejected from the core by the reactor pressure, thereby causing a potentially catastrophic power excursion.

Staff position. Although normal operating procedures for Byron are still under development, procedures for similar Westinghouse designed PWRs require that the reactor be subcritical by a large amount of negative reactivity when the safety rods are inserted. This is accomplished by injecting soluble boron into the reactor coolant system. The reactor is brought critical by first withdrawing all safety rod banks and then by partially withdrawing the control banks as well as controlling boron concentrates. The rods in the control banks are staggered throughout the core and separated by withdrawn safety rods. Thus in the unlikely event that a failed control rod housing impinged against an adjacent safety rod housing, the adjacent safety rod would not be ejected since it would already be fully withdrawn. The rod ejection events analyzed in Byron FSAR section 15.4.8 assume that the most reactive rod in the core is ejected at the most limiting time of core life. Nevertheless, the pressure increase following a rod ejection event would be mild, less than 100 psi and occurs well after the power spike for the event. The reactor system pressure would not exceed the design pressure and therefore would be unlikely to cause failure of additional control or safety rod thimbles. If they did fail, the effect would not be additive since the pressure spike lags the power spike.

A control rod ejection event has never occurred in a PWR in approximately 200 reactor years of operation. The NRC staff believes that the low likelihood of the event, coupled with necessary violation of procedures that would be required for multiple control and safety rods to be inserted when the reactor is not in a highly subcritical state, remove any need to consider multiple rod ejection events in FSARs.

Examples (2) Failure of the main feedwater system followed by a scram system failure, which results in a high level of heat generation in the core of the reactor but low heat removal from the reactor system.

(3) Seizure of a main coolant pump followed by a scram failure.

(4) Continuous withdrawal of control rods with a scram failure.

(5) Loss of electric power to the coolant pump followed by a scram failure.

(6) Loss of turbine steam condenser vacuum with scram failure.

(7) Small coolant pipe rupture with a scram failure.

Staff position. These events are all various types of anticipated transients without scram (ATWS) events. ATWS is the subject of pending rulemaking before the Commission.[2/] The Staff basis for permitting Byron operation pending the outcome of Commission consideration of ATWS is contained in the SER, § 15.6. To further

---

2/ 46 F.R. 57521 (Nov. 24 1981).

reduce the risk from ATWS events, the Staff has required that emergency procedures be developed to assist operators in the recognition of an ATWS event. A discussion of these procedures is contained in the separate affidavit of William Kennedy on contention 4. Due to the low likelihood of ATWS events, combined with the procedural interim actions to be implemented by the Applicant, additional analyses by the applicant are not necessary or required.

Example (8) Large coolant pipe rupture followed by failure of the emergency coolant system to function.

<u>Staff position</u>. The Applicant has analyzed the plant response to a spectrum of large break loss of coolant accidents. As discussed in section 6.3 of the Byron SER, the Staff reviewed the applicant's evaluation model and found it to be in conformance with Appendix K to 10 CFR 50. The fuel damage associated with these events was within the limits of 10 CFR 50.46 and the offsite dose consequences are within the guidelines of 10 CFR 100. Neither Appendix K to 10 CFR 50 nor the General Design Criteria require the Applicant to assume failure of both ECCS trains. The NRC Staff, therefore, does not require that failure of both ECCS trains be evaluated as one of the design basis events in the FSAR. If both ECCS trains were to fail following a large LOCA, the condition would result in inadequate core cooling. Westinghouse has analyzed inadequate core cooling events generically for the purpose of developing defense-in-depth Emergency Response Guidelines. These guidelines will be developed into emergency procedures which will be available to the Byron operators by startup.

Example (9) Spontaneous reactor vessel explosion due to failure of defective closure bolts.

Staff position. The NRC Staff and the ACRS reviewed the effect of reactor vessel head stud bolt failure for McGuire which is a Westinghouse designed plant similar to Byron. The Staff concluded that failure of one stud bolt will not produce sufficient stress on the remaining stud bolts to cause their failure. Although forty-four threaded fastener failures have occurred at various PWR facilities in approximately 200 reactor years of operation, these failures have occurred in steam generator support studs manways, valves, piping supports and reactor vessel internal supports. No failures have occurred in PWR reactor vessel head stud bolts, although two reactor vessel studs failed in a head removal operation at LaCross (a BWR) in 1970. These failures were attributed to water damage and improper heat treatment. The condition was corrected by the licensee. The applicant for Byron complies with the requirements of Regulatory Guide 1.65 for materials and inspection of reactor vessel stud bolts. See FSAR Section A 1.65. Compliance with these requirements makes the occurrence of defective reactor vessel stud bolts at the Byron plant extremely unlikely.

Based on the above discussion, the NRC Staff concludes that multiple head bolt failure to the degree that a reactor vessel explosion could occur is a highly unlikely event which falls outside the single failure criterion of Appendix A to 10 CFR 50 and therefore the event need not be analyzed in FSARs.

Example (10) Errors in regulating the boron chemical concentration in the reactor coolant causing excessive over-power transients or power excursions.

Staff position. Inadvertent boron dilution is one of the design basis events that was evaluated by the Applicant in FSAR Section 15.4.6. The NRC Staff reviewed these analyses and concluded that excessive power excursions will not occur as a result of these events. (See Section 15.2.4.4) The reactor is protected by the reactor protection (scram) system when the reactor is critical. Byron is installing an automatic boron dilution mitigation system for subcritical operation. The system will be designed to terminate dilution and add boron following an excessive increase in startup count rate. Analyses performed by the Los Alamos Scientific Laboratory for the NRC Staff have indicated the power excursion produced by a hypothetical unmitigated boron dilution event from shutdown would be mild (100 MW) and would be automatically terminated by doppler and void reactivity feedback.

Example (11) A large pipe rupture followed by failures of additional pipes and components due to the reaction forces that occur as a result of the pipe rupture.

Staff position. The Staff's review of the dynamic effect associated with high energy piping rupture is discussed in Section 3.6.1 and 3.6.2 of the SER wherein the Staff concluded that the pipe rupture postulation and the associated effects are adequately considered in the plant design for systems inside the containment building, and are therefore acceptable, and meet the requirements of GDC 4.

The NRC staff is still reviewing plant protective features to mitigate pipe whip outside containment. The NRC staff will require that the applicant demonstrate complete compliance with GDC 4 before startup. (SER Section 3.6.1). Analyses of multiple independent high energy piping breaks is not required by the regulations, provided that the applicant demonstrate compliance with GDC 4.

Example (12) Coolant pipe rupture due to a strong pressure surge caused by a core power or under-cooling incident; or a simultaneous rupture of a set of defective control rod drive mechanism housings due to a strong coolant pressure surge, water hammer, or a coolant explosion caused by a molten fuel-water interaction in an accident in which the fuel melts.

Staff position. The Applicant addressed those events which might lead to pressure surges within the primary system in Chapters 5 and 15 of the FSAR. These analyses were reviewed by the NRC Staff as discussed in Chapter 15 and Section 5.2.2 of the SER. Since the reactor system pressure boundary was found to be adequately protected, analyses of additional primary system failure is not required.

Example (13) Steam generator (tube) (sic) rupture.

Staff position. In his deposition on March 12, 1982, Dr. Kaku stated that this example meant steam generator tube rupture rather than steam generator vessel rupture. The Applicant analyzed the consequences of a steam generator tube rupture in Section 15.6.3 of the FSAR. The results were found acceptable to the NRC Staff in Section 15.4.3 of the SER.

The NRC staff has evaluated the effect of failure of one tube on adjacent tubes and does not believe that the failure would be propagated in the time required to achieve safe shutdown even if adjacent tubes were weakened by corrosion. The NRC Staff, therefore, does not require multiple tube failures to be evaluated in FSARs.

Multiple tube rupture events were analyzed by Westinghouse for the purpose of developing Emergency Response Guidelines which will be factored into Emergency Operating Procedures at Byron. Multiple tube ruptures were also performed for the Staff by the Los Alamos National Laboratory. These results indicated that multiple tube failures did not significantly increase the amount of coolant loss to the environment. Even though the break flow would initially be larger, the primary pressure decrease would be more rapid. The break flow would stop when primary system pressure dropped to the secondary system pressure regardless of break size.

Example (14) Improper operator actions in response to a particular accident situation which tends to worsen the accident.

Staff position. Although the improved operator training required by NUREG-0737 reduces the possibility that operators may take incorrect actions, NUREG-0737, Item I.C.1 also requires that plant procedures be developed to include multiple failures which occur as a result of operator error of omission or commission. The Emergency Response Guidelines from which emergency operating procedures for Byron will be developed provide for continuous monitoring of significant plant variables to determine if the correct remedial actions are being taken. The guidelines provide for reinitiation of safety systems which may have been disabled by the operator and provide for continuous checks to determine if a safety system has fulfilled its function. The guidelines are designed so that possible operator misinterpretation of the event would still direct the operator to the correct procedure based on plant conditions. The procedures will direct treatment of the event based on proximity to safety limits if the operator is unable to diagnose the fault directly. Development of the guidelines into Plant Procedures at Byron is discussed in SER Section 13.5.2.3.

The adoption of approved plant procedures and upgraded operator training will further decrease the likelihood of multiple failure events outside of the plant design basis caused by operator error. The NRC Staff therefore does not require them to be analyzed in FSARs.

(15) Accidents caused by faulty gauges and instruments.

Staff position. Westinghouse Emergency Response Guidelines are designed to instruct operators to utilize multiple instrument indications in taking action. Use of multiple instrument indication will minimize the likelihood of the operator taking an incorrect action. If incorrect actions are made, continuously monitoring significant plant variables will permit the operator additional opportunity to rediagnose the event and to take the correct remedial action. Development of these guidelines into Plant Procedures at Byron is discussed in SER Section 13.5.2.3.

The adoption of approved plant procedures and operator training will make multiple failure events which fall outside the plant design basis unlikely at Byron. The NRC Staff therefore does not require them to be analyzed in FSARs.

Dr. Kaku raised three additional examples of multiple failures in his deposition of March 12, 1982. The NRC Staff does not require that the first two be considered in FSARs. The third item has been evaluated in the Byron FSAR by the Applicant. The Staff position on each follows.

Example (16) "No. 1, assume you have hydrogen gas like at Three Mile Island, except you just depressurize the vessel. Assume hydrogen gas inside the reactor. Depressurize the vessel, and will the core be uncovered.

Okay, assuming a variable amount of hydrogen gas, up to 50 percent of zirconium oxidized."[3/]

Staff position. Analyses of transient and accident conditions including LOCA submitted by the applicant in FSAR Section 15 and approved by the NRC staff do not predict that large amounts of hydrogen gas will be produced in the reactor vessel as a result of core damage. The NRC staff evaluated the effect of any noncondensable gas which might form on core cooling in NUREG-0611[4/] and concluded that it would have a negligible effect.

3/ Tr. 123

4/ "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, January 1980.

As a defense-in-depth measure, Byron will be equipped with a reactor vessel head vent system. In the event large amounts of hydrogen gas were produced in the reactor vessel, the operator would open a head vent causing the gas to flow out of the reactor system. Although this action would cause the reactor system to depressurize, opening the vent line to relieve the hydrogen gas would not cause the core to become uncovered, since hydrogen not water would escape from the vent line. Even if the vent were opened for conditions when water would be discharged, the ECCS has sufficient capacity to makeup the loss without core uncovery. Procedures for using the vent will be based on the Emergency Response Guidelines which will be reviewed and approved by the NRC Staff before startup.

Example (17) "The next one would be a feedwater failure and a malfunction of the HPI. Very straightforward. I am assuming full scram. Scram has occurred."5/

Staff position. Both the auxiliary feedwater systems and the HPI (charging pumps) are designed to safety-related criteria in accordance with Appendix A of 10 CFR 50. No single failure of either of these systems would cause loss of function and analysis of this event in FSARs is therefore not required.

For the purpose of developing Emergency Response Guidelines, Westinghouse presented analyses of complete feedwater failure without ECCS in WCAP-9600. These analyses indicate that over one hour is available

---

5/ Tr. 124

to the operator to restore feedwater before core uncovery would occur.

Example (18) "Okay, the next one is a double-ended guillotine break in the cold leg pipe, combined with delayed scram. This, I think, is extremely straightforward. It could almost be done simply by changing one line of the SATAN-6, simply delaying the scram."[6/]

Staff position. The NRC staff does not require ATWS events to be evaluated in FSARs for the reasons stated in the SER and above. However, in the case of a large break LOCA, the Staff is concerned that the dynamic forces of the event might delay insertion of the safety rods. For this reason, the large break LOCA evaluations for Byron were performed without taking credit for scram at any time. See WCAP-8334.[7/] Delayed scram has little effect on the course of the event since void formation in the core adds a large amount of negative reactivity early in the event. After blowdown the core would be reflooded with borated water from the accumulators and the refueling water storage tank which would preclude recriticality. This event is already considered in LOCA analysis in the Byron FSAR and, therefore, no further analyses are required.

6/ Ibid.

7/ "Westinghouse Emergency Core Cooling System Evaluation Model-Summary," WCAP-8339, June 1974.

ATTACHMENT B

Design Basis Events*

Decrease in Feedwater Temperature (15.1.1) (15.2.2)

Increase in Feedwater Flow (15.1.2) (15.2.2)

Steam Pressure Regulator Malfunction Resulting in Increasing Steam Flow (15.1.3) (15.2.2)

Inadvertent Opening of a Steam Generator Atmospheric Dump or Safety Valve (15.1.4) (15.2.2)

Steam Pressure Regulator Malfunction Resulting in Decreasing Steam Flow (15.2.1)*

Turbine Trip (15.2.3) (15.2.1)

Loss of Electrical Load (15.2.2) (15.2.1)

Inadvertent Closure of Main Steam Line Isolation Valve (15.2.4) (**)

Loss of Condenser Vacuum (15.2.5) (**)

Loss of Nonemergency Alternating Current Power (Offsite) to the Station Auxiliaries (15.2.6) (15.2.1)

Loss of Normal Feedwater Flow (15.2.7) (15.2.1)

Uncontrolled Control Rod Group Withdrawal From Subcritical Condition (15.4.1) (15.2.4.1)

Uncontrolled Control Rod Group Withdrawal at Power (15.4.2) (15.2.4.2)

Control Rod Assembly Drop (15.4.3) (15.2.4.3)

Startup of Inactive Reactor Coolant Pumps (15.4.4) (15.2.2)

Chemical Addition System Malfunction (15.4.6) (15.2.4.4)

Inadvertent Operation of ECCS During Power Operation (15.5.1) (**)

Inadvertent Opening of a Pressurizer Safety or Relief Valve (15.6.1) (**)

---

* The parenthetical references refer to FSAR and SER Sections, respectively.

** Non-bounding events reviewed by the Staff but not specificly discussed in the SER.

Single Control Rod Assembly Withdrawal (15.4.3) (15.2 4.3)

Loss of Reactor Coolant Flow (15.3.1 & 15.3.2) (15.2.1)

Steam Line Break (15.1.5) (15.3.4)

Feedwater Piping Break (15.2.8) (15.3.5)

Reactor Coolant Pump Shaft Seizure (15.3.3) (15.3.6)

Reactor Coolant Pump Shaft Break (15.3.4) (15.3.6)

Rod Ejection (15.4.8) (15.3.2)

Break In Instrument Line or Reactor Coolant System Line Penetrating Containment (15.6.2) (15.4.6)

Steam Generator Tube Rupture (15.6.3) (15.4.3)

Loss of Coolant Accident (15.6.5) (6.3)

Fuel Misloading (15.4.7) (15.3.1)

WALTON L. JENSEN, JR.

PROFESSIONAL QUALIFICATIONS

I am a Senior Nuclear Engineer in the Reactor Systems Branch of the Nuclear Regulatory Commission. In this position I am responsible for the technical analysis and evaluation of the public health and safety aspects of reactor systems.

From June 1979 to December 1979, I was assigned to the Bulletins and Orders Task Force of the Nuclear Regulatory Commission. I participated in the preparation of NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants."

From 1972 to 1976, I was assigned to the Containment Systems Branch of the NRC/AEC, and from 1976 to 1979, I was assigned to the Analysis Branch of the NRC. In these positions I was responsible for the development and evaluation of computer programs and techniques to calculate the reactor system and containment system response to postulated loss-of-coolant accidents.

From 1967 to 1972, I was employed by the Babcock and Wilcox Company at Lynchburg, Virginia. There I was lead engineer for the development of loss-of-coolant computer programs and the qualification of these programs by comparison with experimental data.

From 1963 to 1967, I was employed by the Atomic Energy Commission in the Division of Reactor Licensing. I assisted in the safety reviews of large power reactors, and I led the reviews of several small research reactors.

I received an M.S. degree in Nuclear Engineering at the Catholic University of America in 1968 and a B.S. degree in Nuclear Engineering at Mississippi State University in 1963.

I am a graduate of the Oak Ridge School for Reactor Technology, 1963-1964.

I am a member of the American Nuclear Society.

I am the author of three scientific papers dealing with the response of B&W reactors to Loss-of-Coolant Accidents and have authored one scientific paper dealing with containment analysis.