April 22, 2003

Ms. Joanne Steele, Director
Oconee Project
Action for a Clean Environment
319 Wynn Lake Circle
Alto, GA 30510-5218

Dear Ms. Steele:

I am responding to your March 4, 2003, letter to Mr. Steven D. Bloom. In your letter you asked eight specific questions and made other statements that I will be responding to.

Question 1: Why do you consider it ethical to "jump the gun" in your words, and post a document that rates the 3 worst reactor heads in the country as low susceptibility to deterioration, especially unit 3 of Oconee that is still in operation, BEFORE replacements are done? I've seen no posting of a correction. This gives a false sense of security of the present situation.

Answer: Our web site on Reactor Vessel Head Degradation lists Oconee, Units 1 and 2, as high susceptibility plants. Oconee, Unit 3, had originally been listed as a low susceptibility plant, based on Oconee management's plans to replace the reactor pressure vessel head (RPVH) during the next outage. Rather than continuing to reflect the future status, following the outage, we updated the web site on March 5, 2003, to show Oconee, Unit 3s current status as a high susceptibility plant.

During the February 24, 2003, meeting, the NRC staff member incorrectly thought that Oconee, Unit 3 was already shutdown and would be replacing its RPVH during the outage. His comments were based on that incorrect assumption. Oconee, Unit 3 has not yet entered its outage, but the licensee still plans to replace its RPVH when it does enter the outage this spring. We trust that changing the web site status from low to high susceptibility for Oconee, Unit 3, to reflect the current status, is responsive to your concern.

Question 2: What if simultaneous rod ejections occur, and there is tearing of insulation during that accident, and clogging of the sump from debris?

Answer: The simultaneous ejection of multiple control rods is considered to be a very unlikely event and, hence, is not within the design basis of domestic reactors. Reactors are designed for the reactivity excursion due to the ejection of the control rod with the greatest reactivity worth. Reactors are also designed to mitigate a spectrum of pipe rupture sizes that envelopes the size of the rupture expected from a single control rod ejection. Should two or more control rods be ejected simultaneously, the initial response of the containment and reactor coolant system would be similarly enveloped by that of analyzed pipe breaks, which are generally up to a maximum total break area equal to the double-ended rupture of the largest pipe in the reactor coolant system.

If a rupture of the reactor coolant system pressure boundary occurred under normal operating conditions, significant damage to insulation and other materials would be expected in the vicinity of the rupture. The degree of damage expected depends upon a number of parameters, including the size of the rupture and the sturdiness of the nearby insulation. At most plants, control rod ejections would be expected to generate a substantially smaller quantity of debris than postulated ruptures on the main reactor coolant system piping. This is because the reactor vessel is isolated from most potential debris sources (other than its own insulation) and the rupture size of control rod ejections is expected to be smaller. Consequently, control rod ejections are generally not the most limiting challenge with respect to sump clogging. Considering this comparatively less severe consequence in conjunction with the very low likelihood of the event occurring, the simultaneous ejection of multiple control rods is not expected to contribute significantly to the risk associated with sump clogging.

Furthermore, in the event of a rod ejection the residual heat removal system is available to be operated in the shutdown cooling mode to cool the reactor core without the need to take suction from the recirculation sump. Therefore, even if the recirculation sump were to clog with debris, plant operators will be able to mitigate a control rod ejection event using shutdown cooling, once it is understood that the reactor coolant system piping is intact and that the location of the leakage is from an ejected rod.

Question 3: How will the reactor be shut down if there is damage and leaking at a fast rate?

Answer: For larger breaks in the reactor coolant system, pressurized water reactors (PWRs) typically would not need control rods to reach shutdown conditions. The initial loss of water from the break causes the water in the core to flash to steam (voiding). Since sustainable fission reactions in PWRs require water for neutron moderation, the voiding causes the termination of the reaction. Additionally, all PWRs have safety injection systems that act upon a loss-of-coolant accident to inject borated water into the reactor. The water acts as a cooling mechanism for the nuclear fuel, and the boron in the water absorbs neutrons to keep the reaction shut down.

However, rod ejection accidents typically do not follow the above process of shutting down. Following a multiple rod ejection accident, depending upon the location and reactivity worth of the ejected rods, the reactor may not immediately shutdown (reach a subcritical condition) when the remaining, undamaged control rods drop into the core. But, a process called doppler feedback in the fuel, a negative moderator temperature coefficient of the reactor coolant, and localized voiding of the moderator will limit the severity of the reactivity excursion. Also, the reactivity spike caused by the rod ejections will cause the safety injection system to initiate. As mentioned above, injection of the borated water from the safety injection system will in turn ensure that the reactor reaches shutdown.

Question 4: Has Davis-Besse's head been replaced yet?

Answer: Yes, Davis-Besse purchased a replacement head from the canceled Midland plant and has made some minor modifications to allow it to be used. Davis-Besse has not yet restarted from their outage. They are planning on restarting in the Spring of 2003, with a replaced RPVH.

Question 5: Is the North Anna reactor that had 49 serious cracks remaining off line until vessel head replacement can be done?

Answer: The North Anna, Unit 2 reactor remained off line until the head was replaced and restarted at the end of January 2003. North Anna, Unit 1 recently restarted following its refueling outage and head replacement. None of the indications identified in the North Anna, Unit 2, head were a challenge to the structural integrity of the head and we did not identify significant safety concerns. The indications on the North Anna, Unit 2, RPVH were required to be fixed before the unit could be restarted, which was accomplished by the vessel head replacement.

Question 6: In the meeting there was mention of unidentified leaks of up to one gallon per minute from the primary systems that may drip onto the vessel head or seams. This could add up to 1,440 gallons a day or 43,000 gallons in a 30 day month. Is this common? You said it may be important to identify these leaks. I would think so!

Answer: Unidentified leak rates of one gallon per minute are not common. To our knowledge there has never been that kind of an unidentified reactor coolant leak rate that has leaked onto the vessel or vessel head. There is a 1-gallon per minute technical specification limit for unidentified leakage at PWRs. If this leak rate is reached they must shut down. There is a zero gallon per minute limit for reactor coolant pressure boundary leakage, such as from cracks. If any know pressure boundary leakage occurs, the plant would have to shut down.

Generally, the unidentified leak rates at PWRs are very small and usually associated with valve stem leakage or control rod drive mechanism (CRDM) flange or reactor coolant pump seal leakage. At most PWRs, the actual unidentified leak rate during normal plant operation is usually in the one-to two-tenths of a gallon per minute range. The CRDM and reactor vessel level instrumentation tubing flanges are basically the only source of leakage above the reactor vessel head. Observed leak rates from CRDM flanges have generally been very small and do not usually approach 1-gallon per minute.

Question 7: How are the CRDMs monitored when the reactors are in operation?

Answer: Neither the CRDM flanges or nozzles are specifically monitored for cracking or leakage during normal operation. At most plants, the nozzles can only be monitored or inspected for cracks while the unit is shutdown and the reactor vessel head removed. With regards to leakage monitoring, there are leak detection systems in place that monitor for reactor coolant system leakage from all sources within the reactor containment, including possible leakage from the CRDM nozzles and flanges. However, the methods and equipment used to monitor unidentified leakage is not sensitive enough to detect the very small leak rates typically attributed to these sources.

Question 8: How is the vessel itself tested for strength and integrity? These reactors are so old, and the metal is bound to deteriorate all around. We believe total inspection of the entire vessel should be done, top to bottom, inside and out. Is that possible?

Answer: NRC and the licensee ensure the continued integrity of the vessel throughout its operating life through multiple requirements, monitoring programs, and inspections. For example, the reactor pressure vessel and its associated piping undergoes a pressure hydro consistent with the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Requirements are in place for minimizing the chance of and monitoring for the effects of degradation caused by various types of corrosion, including neutron irradiation embrittlement, boric acid corrosion, and general corrosion. First, reactor vessels are designed and built to specific technical requirements (e.g., ASME Boiler and Pressure Vessel Code) that take into account the potential affects of corrosion, to ensure the integrity of the plant over its entire operating life. As an example of this, the inside of the vessel wall is clad with stainless steel to inhibit corrosive attack of the ferritic material. Second, over the life of the plant, controls are placed on the operating environments to minimize corrosion. As an example of this, primary water chemistry controls are in place to limit exposure of the reactor vessel to detrimental chemicals, such as chlorides and sulfides. Third, extensive inspections and monitoring of the reactor vessel materials are routinely conducted over the life of the plant to ensure no active degradation is taking place. Inspections include those detailed in ASME Code, Section XI, that are required by 10 CFR 50.55a. Monitoring of the reactor pressure vessel materials is mandated by 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” and Appendix H, “Reactor Vessel Material Surveillance Program Requirements.” In summary, a layered regulatory program involving requirements, monitoring, and inspections exists to ensure the continued integrity of the vessel throughout its operating life.

In addition to the above questions, your letter contained statements that were not identified as questions by you, but that I would like to respond to. In your letter you stated:

> Oconee reactors were relicensed before all the concerns of aging reactors were fully taken into consideration. We recommend that these and other reactors

> being considered for relicensing be held to their original licensed operation. Problems will continue to arise with the vessels. Nuclear accidents would be catastrophic and impossible to clean up. Financial resources don't exist to meet emergency response to serious accident, nor can emergency plans prevent the resulting high death and cancer rates a radioactive released from a damaged reactor.

License renewal is based on the determination that the NRC's regulatory process is adequate to ensure that current operating nuclear power plants maintain an adequate level of safety and over the life of a plant, this level of safety is enhanced as new information or operating experience is gained. The license renewal review is focused on plant systems, structures, and components for which current activities and requirements may not be sufficient to manage aging effects during the period of extended operation (i.e., 40 to 60 years). Currently, there are extensive requirements for monitoring, surveillance, detection, and preventive maintenance to ensure that a nuclear power plant is operated safely and that its current licensing basis provides an adequate level of safety. After license renewal, this current licensing basis must be maintained in the period of extended operation in the same manner and to the same extent as during the original licensing term. Additional programs or activities may be required for the period of extended operation as a result of the license renewal review to ensure that the effects of aging on the functionality of systems, structures, and components will be managed to maintain the current licensing basis.

The NRC has determined that issues concerning operation during the currently licensed term (first 40 years) must be addressed as part of the current operating license and cannot be deferred to a renewal review. Your concerns regarding reactor head vessel degradation involve the safe operation of nuclear power plants currently operating and these concerns are being addressed by the NRC and licensees. Any requirements resulting from the resolution of this issue becomes part of a plant's licensing basis, that must be carried forward and complied with in the period of extended operation.

As for the financial resources related to emergency response, the Price-Anderson Act specifically exists for this purpose. The Act provides a system to pay funds for claims by members of the public, including emergency assistance claims, for personal injury and property damage resulting from a nuclear incident. Price-Anderson entails a two-part insurance system for liability payments. The first consists of primary nuclear liability insurance whereby utilities operating large power reactors pay a premium each year for a fixed amount of liability coverage, currently $300 million. In the event of a nuclear incident causing damages exceeding $300 million, each large licensed nuclear power reactor would be assessed a retrospective premium of up to $83.9 million per reactor per accident. Under this system, the total funds available to pay claims for an incident would exceed $9 billion. At the time of the Three Mile Island accident, the insurance companies were on the scene very rapidly to begin writing checks for emergency claims, including temporary housing, food, and other expenses. Significantly increased funds presently exist to meet similar emergency response needs.

Emergency planning is one tier in the NRC's "defense in depth" safety philosophy. The overall objective of emergency response planning is to assure that, in the event of an accident at the facility, radiation doses to persons off site would be below doses that could result in acute or long term health effects (e.g. prompt fatalities or subsequent cancers).

For planning purposes, we define two planning zones around nuclear power plant sites. The first is an emergency planning zone covering an area of about 10 miles in all directions around nuclear power plants where the greatest potential for radiological effects from a release exists. Protective actions for members of the public in this zone could involve evacuation or sheltering. Consideration of these protective actions is prompted at very low projected dose levels. They are also doses far below a level that would lead to long-term or appreciable health effects. A second, extended planning zone of about 50 miles is also established around each plant to deal with potential lower-level, long-term risks primarily due to exposure from ingestion of contaminated food and water.

Emergency planning is a dynamic process. Emergency response plans are tested in frequent small-scale drills and periodic full-scale emergency exercises that simulate serious reactor accidents. The plans and their implementation are periodically reviewed to confirm that they are being adequately maintained and address changing circumstances appropriate to any given site.

I hope that this answers your questions. If you have additional questions please do not hesitate to contact me at 301-415-3037 or Mr. Steven D. Bloom at 301-415-1313.

Sincerely,

/RA/

Scott W. Moore, Acting Director
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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