ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

August 10, 1971

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D.C. 20545

Subject: REPORT ON NEWBOLD ISLAND NUCLEAR GENERATING STATION UNITS NOS. 1 AND 2

Dear Dr. Seaborg:

At its 136th meeting, August 5-7, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application by the Public Service Electric and Gas Company for a permit to construct the duel-unit Newbold Island Nuclear Generating Station. This project was also considered at the 130th, 133rd, 134th, and 135th meetings of the Committee on February 4-6, May 6-8, June 10-12, and July 8-10, 1971, respectively; and at Subcommittee meetings on June 3, 1970 at Argonne National Laboratory, and on February 3, March 29, April 26, June 3, July 7, and August 4, 1971 in Washington, D. C. During its review the Committee had the benefit of discussions with representatives and consultants of the applicant, the General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below. The Committee reported the results of its pre-application site review to you in a letter dated September 10, 1969.

The station will be located in New Jersey on 530-acre Newbold Island which is near the east bank of the Delaware River about 4-1/2 miles south of Trenton, New Jersey (1970 population - 105,000) and 11 miles northeast of Philadelphia, Pennsylvania (1970 population - 2,000,000). The nearest population center is a grouping of suburbs in Bucks County, Pennsylvania, known collectively as Levittown (1970 population - 72,000), with its nearest boundary 3.4 miles from the site. The applicant has specified a radius of one mile for the low population zone, which had in 1969 a transient population associated with industry of approximately 1200, and a small resident population which is expected to be about 100 by 1985. The minimum exclusion distance is 700 meters, which extends to the west bank of the Delaware River. As pointed out in the Committee's report of September 10, 1969, a relatively high population density is associated with this site.

Each unit includes a boiling water reactor to be operated at 3293 MWt. With respect to core design, power level, and other features of the nuclear steam supply system, the Newbold Island units are essentially duplicates of the Browns Ferry Units 1, 2 and 3, and Peach Bottom Units 2 and 3. Waste heat from the station will be rejected to the atmosphere by natural draft cooling towers.

In its report of September 10, 1969, the Committee listed several matters which it believed warranted special attention in the design of a plant for the Newbold Island site. In response to these recommendations, the applicant has included in the Newbold Island design several features, in addition to those normally provided for boiling water reactor units, to reduce still further the potential for release of radioactivity to the environment. The principal additional features are described below:

Reactor Building. For each unit, the conventional steel drywell and suppression chamber primary containment, the fuel handling area and spent fuel pool, and the principal components of the engineered safety features are contained in an unlined reinforced concrete building of cylindrical shape with a domed roof. This building is designed to Class I seismic standards and to resist the standard tornado, and missiles from this or other sources. The building can resist an internal pressure of 2 psig, and inleakage at a differential pressure of 1/4-inch of water will be limited to 10 percent of the building volume per day. A filtration, recirculation, and ventilation system (FRVS) is provided to recirculate and filter the reactor building atmosphere and maintain the building at a negative pressure relative to the outside environment.

Main Steam Lines. A low-leakage, slow-acting, stop valve has been added downstream of the two fast-acting valves in each main steam line, and a seal air system has been provided to further reduce leakage of radioactivity after main steam line isolation. The portion of the main steam lines containing the isolation valves is enclosed in a Seismic Class I tunnel chamber connected to the reactor building so that any out-leakage following the unlikely event of a design basis loss-of-coolant accident will be treated by the reactor building FRVS before release to the atmosphere. The entire length of the main steam lines up to and including the turbine stop valve will be designed to Class I seismic standards. The main steam lines from the third isolation valve to the turbine stop valve will be designed and fabricated in substantial accordance with the requirements for AEC quality assurance Classification Group B. In addition, selective inspection of critical areas of this piping will be performed during refueling outages.

Radioactive Waste Disposal. The radioactive waste disposal systems include several features beyond those normally provided in boiling water reactor plants. The liquid waste system permits the recycling of equipment and floor drain wastes and the evaporation of chemical and laundry wastes before discharge to the environment. The gaseous waste system provides for the recombining of hydrogen and oxygen, condensing the vapor, hold-up for decay of short-leved isotopes, and cryogenic separation of the noble gases. Krypton and xenon may be stored for periods sufficiently long that krypton-85 becomes the only significant remaining radioisotope. Provisions will be made to utilize non-radioactive steam in the turbine gland seals and to process containment purge gases when deinerting. The Committee believes that these waste management systems are capable of limiting releases of radioactivity to the environment to levels that are as low as practicable.

Reactor Vessel Integrity. The applicant has described improvements in the design and fabrication of the reactor vessel. These include redesign of the large nozzles to reduce stress concentrations; redesign of the bottom head to reduce the number of welds and improve the capability for in-service inspection; and improved procedures and standards for inspection during fabrication. The applicant has studied the problems related to possible degradation of reactor vessel integrity and has concluded that a nozzle failure or a small break would not impair the integrity of the biological shield, the primary containment, or the reactor internals, and would not affect the ability to cool the core. In addition, the biological shield has been redesigned to increase substantially its ability to withstand internal pressures, jet forces, or missiles.

Emergency Core Cooling System. The emergency core cooling system (ECCS) has been modified in two ways. The high-pressure coolant injection (HPCI) system has been changed to inject water directly to the core through the core spray sparger rather than into the downcomer region via the feedwater sparger. In addition, the applicant has stated that the steam-turbine driven HPCI pump will be modified to the extent feasible to increase the volume of water delivered to the core. The low-pressure coolant injection (LPCI) system has been changed to inject water inside the core shroud through four separate vessel penetrations, rather than through the recirculation lines. The applicant has stated that these changes provide increased reliability of these systems and reductions in the peak clad temperatures attained in the unlikely event of a loss-of-coolant accident.

The Committee believes that the design changes described above are suitably responsive to the concerns stated in its letter of September 10, 1969 regarding additional matters which should be considered for a plant at the Newbold Island site.

In the event of an unisolable break of an instrument line or a process line, reactor coolant will be discharged to the reactor building. Since the instrument lines will contain a 3/8-inch flow-restricting orifice inside the primary containment, failure of as many as eight such lines will not lead to pressures inside the reactor building greater than the 2 psig at which it relieves to the environment. However, failure of a process line, if not isolated in a very short time, could lead to pressures in excess of this relief pressure and significant amounts of reactor coolant would be discharged to the environment. Although the off-site doses from such an accident would be well within the 10 CFR Part 100 guidelines, they would be comparable to or greater than the doses calculated for other less probable accidents. The Committee believes, therefore, that the applicant should make design provisions for reducing the quantity of reactor coolant discharged to the reactor building in the event of a process line break.

The applicant has studied design features to make tolerable the consequences of failure to scram during anticipated transients, and has concluded that automatic tripping of the recirculation pumps and injection of boron could provide a suitable backup to the control rod system for this type of event. The Committee believes that this recirculation pump trip represents a substantial improvement and should be provided for the Newbold Island reactors. However, further evaluation of the sufficiency of this approach and the specific means of implementing the proposed pump trip should be made. This matter should be resolved in a manner satisfactory to the AEC Regulatory Staff and the ACRS during construction of the plant.

The applicant has stated that a system will be provided to control the concentration of hydrogen in the primary containment that might follow in the unlikely event of a loss-of-coolant accident. The proposed system is not capable of coping with hydrogen generation rates in accordance with current AEC criteria unless the primary containment is inerted. Therefore, the Committee believes that the containment should be inerted and that the hydrogen control system should be designed to maintain the hydrogen concentration within acceptable limits using the assumptions listed in AEC Safety Guide 7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident."

Other problems related to large water reactors have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports. The Committee believes that resolution of these items should apply equally to the Newbold Island Station.

The Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items, the Newbold Island Nuclear Generating Station Units Nos. 1 and 2 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Additional comments by Dr. H. O. Monson, Dr. D. Okrent and Dean N. J. Palladino are attached.

Sincerely yours,

/s/

Spencer H. Bush
Chairman

References - Newbold Island Nuclear Generating Station Units Nos. 1 and 2

1. Public Service Electric and Gas Company letter dated February 27, 1970; License Application; Preliminary Safety Analysis Report (PSAR), Volumes 1 through 5

2. Amendments Nos. 1 through 5 and Nos. 7 through 9 to PSAR

ADDITIONAL COMMENTS BY DR. H. O. MONSON, DR. D. OKRENT AND DEAN N. J. PALLADINO

Although the large, low pressure, high in-leakage secondary reactor building proposed by the applicant for Newbold Island Units 1 and 2 represents an improvement over reactor buildings currently employed for BWRs at sites with lower surrounding population densities, we believe that further improvement is appropriate. The relatively small volume of the steel pressure-suppression type primary containment introduces some crowding of equipment and some attendant problems in the simultaneous accomplishment of full protection against violation of primary containment by possible missiles, jet forces, and pipe whip, and accomplishment of full access for in-service inspection. Some further protection would be provided against extremely low-probability accidents involving a concurrent loss of primary system integrity and a limited violation of primary containment by the use of a large, relatively high-pressure (of the order of 10 psi, as has been proposed for a BWR at another site having a comparable surrounding population density), low-leakage, secondary containment building. Such a high-pressure, secondary containment, coupled with a pressure-suppression primary containment, provides a combination which can tolerate a fairly substantial violation of primary containment arising from the same event which caused a loss of coolant, as well as further protection against unforeseen events. We believe that this improvement in safety capability is warranted for a more densely populated site like Newbold Island, and recommend that the issuance of a construction permit be contingent on the use of a high-pressure, low-leakage secondary containment.

For postulated loss-of-coolant accidents involving small break sizes, the high-pressure coolant injection system (HPCI) arranged so as to inject into one of the core spray loops is predicted by the applicant to be highly effective in limiting peak clad temperatures to moderate levels. We believe that for a high power, high-power-density reactor at a site as densely populated as Newbold Island, the applicant should give further consideration to the use of an HPCI system on the second core spray loop. The purpose would be to provide redundancy of this means of protection in the event that the single HPCI system became ineffective because of failure of an HPCI component or because the accident arose from rupture of the core spray line into which the HPCI injects. The automatic depressurization system which together with the low-pressure emergency cooling systems constitutes an alternate means for coping with small breaks, albeit by introducing a larger opening, would continue to serve as a backup.