STATEMENT BY MARY SINCLAIR
before the Advisory Committee on Reactor Safeguards
May 20, 21, 1982

In the first summary statement that the Advisory Committee on Reactor Safeguards (ACRS) prepared for the Midland nuclear plants dated February 6-8, 1969, a number of significant issues were raised about which I believe the people of this area need explanation and some knowledge of their disposition. This memorandum was supressed at the time and was never produced during the construction license hearings when citizens asked for information under discovery.

The Committee expressed a great deal of concern about the high population within four miles of the plant site, as well as the minimal engineering safeguards that were proposed in the application and the use of less than conservative assumptions in the dose calculations.

The Committee recommended that: (1) The facility should be equipped with adequate engineered safety features and protective systems; (2) the facility should be designed sufficiently conservatively - particularly in respect to determination of exclusion area and low population zone; assurance should be provided of low potential doses at short distances from the reactor in the unlikely event of a serious accident; evaluation made of the number and location of people who could be safely and quickly evacuated in such an event; and, use of conservative assumptions, for example, for those related to meteorology, in dose calculations; (3) the facility should be designed, constructed and utilized sufficiently conservatively; and (4) the facility should be provided with thoroughly structured, effective emergency plans, including evacuation plans.

Other important issues that were a part of this original letter included the statement that the containment of these reactors was designed with a leakage rate that was greater than for most other reactors of this type. The Committee expressed some questions as to the suitability of B&W reactors for marginal sites of this type. They were concerned about the protection required against reactor vessel splits and cavity flooding systems. This ACRS Committee was the first to question the use of process steam in products to be consumed by people.

We believe that the population here is entitled to a clear and complete explanation of how these original questions have been met and resolved since we were prevented, because of supression of this document, from examining these issues in the construction license hearing.

Also, in 1969 when these reactors were being designed, the ACRS stated in a letter to Dr. Glen Seaborg who was then chairman of the ACRS that there was urgent need for additional research and development in a number of significant areas for the kind of large-sized reactors similiar to the Midland nuclear plant that were being planned for construction at that time.

These areas included, gaining an understanding of modes and mechanisms of fuel failure, the possible propagation of fuel failure, the generation of locally high pressures if hot fuel and coolant are mixed and gaining an understanding of the various mechanisms of potential importance in describing the course of events following partial or large scale core melting. It is important for this Committee to describe for the public of this area to what extent all these areas, which urgently needed research attention according to the ACRS, have been met at the Midland nuclear plant.

We do know now that 10 years after these recommendations were made that the Three Mile Island accident demonstrated dramatically the fact that little was known by the Nuclear Regulatory Commission about the course of events following partial or large scale core melting. We need to know how many of these other areas have been similarly neglected. Since these plants are similar to the one at the Three Mile Island accident, the public needs assurance that all the lessons learned at Three Mile Island have been carefully evaluated and incorporated in the final construction of these plants.

The kinds of design deficiencies of the B&W plants that were demonstrated by the Three Mile Island accident must be shown to have adequate and conservative engineered compensations or this Committee should recommend lower power output.

The electrical qualification deficiencies that were found in the numerous critical safety systems at Three Mile Island should all be reviewed and resolved at this plant. These include systems relating to: Core Flood, Containment Spray, Emergency Core Cooling, Auxiliary Feedwater, Nuclear Service Water, Containment Isolation, Decay Heat Removal, and Containment Cooling.

The consequences of replacing the steam generators, which is considered inevitable during the lifetime of pressurized water reactors, should be evaluated for the high radiation dose such an operation can inflict on nearby populations and workers within the Dow plant who are actually within the exclusion area.

This ACRS meeting has been scheduled while the soil settlement hearings are still underway. In fact, the licensing board has halted construction in some areas because even the proper procedures for quality assurance for underpinning remedial work has not been agreed upon.

By meeting at this time for a final safety review, this ACRS Committee is prejudicing the findings of the licensing board's review on what is regarded as one of the most serious safety and quality control problems in the nuclear industry in the country.

Finally, it tests the limits of my credibility to believe that this Committee has been able to read, assimilate and evaluate the significance of the 800-page SER Report that was issued only one week ago as the final safety evaluation for these plants.

Mary P. Sinclair

Mary P. Sinclair, M.S.