NRC PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 22 1979

Mr. Gary Ciurczak
711 Parker Blvd., #7
Buffalo, New York 14223

Dear Mr. Ciurczak:

This is in response to your letter of July 4, 1979 to Dr. Joseph Hendrie requesting information on several items relating to nuclear power safety. Your questions are answered in the order in which they appear in your letter.

Strontium-89 and -90 analyses have been deleted as requirements for monitoring in environmental samples (outside the plant boundary), but are required to be monitored in the radioactive effluents from the plant. The decision to omit the sampling of strontium in the environment was based upon the following: (a) Data gathered from over 200 plant-years of operating experience have shown that nuclear plant related strontium was being detected at extremely low levels or not at all; (b) the strontium monitoring program costs a minimum of $7,500. per station per year; (c) Because of the low amounts of strontium released from nuclear power plants, they can be detected more readily in the effluents (before release to the environment) than in the environment (outside the plant boundary and dispersed in the atmosphere and hydrosphere); and (d) Cesium-137 is a good indicator for the presence of strontium-89 and -90, is routinely monitored in the environment, and is more easily detected.

Cesium-137 and strontium-90 are both fission products with similar fission yields and radioactive half lives, but cesium-137 has a higher escape rate coefficient from the fuel rod to the primary coolant. Demineralizers in the primary coolant and radwaste treatment systems remove more strontium than cesium due to the greater ion exchange property of strontium. However, once the cesium-137 and a smaller amount of strontium-90 are in the environment, strontium accumulates in aquatic organisms more readily than cesium. The net result of the above-mentioned effects is that the presence of cesium-137 is expected to be larger than that of strontium-90 by a factor of 1000-100,000 in water and by a factor of 20-2000 in aquatic organisms.

Strontium-89 and -90 are beta emitters and require a chemical separation procedure which is lengthy and time consuming. Cesium-137's activity is much more easily determined by gamma-ray spectrometry.

7909250604

1023 138

Cesium-137 is considered an earlier and better indicator for the presence of strontium than the strontium itself, which must be detected at the very low levels expected from reactors. This is because the cesium production is associated with strontium-90, and its (cesium) changing levels can be determined earlier than by analyzing for strontium. There should be no cases by which release of strontium to the environment would not be accompanied by a release of cesium.

Your statement that NRC considers only external gamma radiation doses received from passing clouds of radioactive gases is reported is incorrect. The NRC considers all important pathways such as the ingestion of fish, milk, water and vegetation, inhalation, and direct radiation from the ground, from all forms of radioactive effluents from the plant and the associated total body and critical organ doses. The NRC limits the radioactive effluents from the plants to ensure the safety and health of the population through regulations which are part of the plant's license to operate. Nuclear power facilities are routinely inspected for compliance with these regulations by the Office of Inspection and Enforcement.

Your concern for the monitoring of effluents at the Three Mile Island facility is based on the belief that the only instruments available for radioactive analyses were thermoluminescent dosimeters. Actually these dosimeters are but one part of the environmental monitoring program. Routine environmental monitoring of Three Mile Island has included (in addition to TLD's) air sampling, precipitation, milk, green leafy vegetables, river water, drinking water, sediment, fish, and aquatic vegetation. The determination that xenon and iodine were the principal radionuclides released was determined by gamma spectrographic measurements in the environment, in the contents of the waste gas tanks, of the gases in the containment building, and the actual gas released to the environment, not by reading the TLD's.

In the third paragraph of your letter, you suggest that the generation of hydrogen by the reactor core was anticipated by the AEC, but not dealt with adequately in the licensing of Three Mile Island. It is true that the possibility of hydrogen production has been understood for many years. For example, Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) addresses the generation of hydrogen in a reactor core during a loss of coolant accident. Specifically, Item 5 of Section A of Appendix K "ECCS Evaluation Models" of 10 CFR Part 50 discusses "Metal-Water Reaction Rate". The purpose of Item 5 is to establish conservative rates for the generation of hydrogen from metal-water reactions by an overheated reactor core during a loss of coolant accident. The NRC has also had requirements associated with the control of hydrogen inside containment (10 CFR 50.44). These measures were intended for hydrogen mitigation in the event of an overheated core. However, the appearance of non-condensable gasses, like hydrogen, within the reactor pressure vessel during the course of an accident was not anticipated in the manner in which it occurred at TMI-2. The NRC requirements did not anticipate the sequence of events and specific circumstances associated with the presence of the hydrogen gas in that accident.

The NRC staff has learned a great deal from the accident. We expect to learn more as our studies continue. The methods of generation and control of non-condensable gasses within the reactor cooling system are receiving considerable attention at this time.

It is disturbing to hear that you feel that the Nuclear Regulatory Commission is primarily concerned with protecting the nuclear industry rather than determining its safety for our country. We want to emphasize that our goals and policies are directed to assuring that the health and safety of the public have been and will be protected from the impacts associated with the use of nuclear power to generate electricity for our citizens. Attached is a copy of the Code of Federal Regulations, Title 10, Energy. 10 CFR Parts 20, 21, 50 and 51 should be of particular interest to you as they pertain more directly to nuclear power plant operation. In particular, Appendix I to 10 CFR Part 50 defines "as low as reasonably achievable (ALARA)" to protect the public from routine radioactive releases from nuclear power plants. Also attached is NUREG-0472, a document which describes the implementation of the ALARA requirements. These requirements serve as one portion of the license to operate a power plant and are provided as an example of the manner in which these regulations are implemented.

All applications for construction and operation of nuclear power plants are reviewed in detail by the NRC staff. Attached is a seven page paper entitled "The Reactor Licensing Process" which explains in detail the licensing process. As mentioned in this paper, some meetings are open to the public, and the hearing process conducted by the Atomic Safety and Licensing Board provides for public interaction in the licensing process.

We currently are investigating previously licensed nuclear power plants as a result of lessons learned from Three Mile Island-2 and as part of our ongoing program to systematically evaluate licensed plants. The results of these reviews will be publically available.

As new standards are developed for the regulation of nuclear power, they are published in draft form in the Federal Register. These drafts provide for public comments and discuss in detail the sources used in establishing the standard and the contact within the NRC to whom to direct inquiries.

We hope this information will help you better understand the role of the NRC in the licensing process of nuclear power plants.

Sincerely,

Daniel R. Muller

Daniel R. Muller, Acting Director
Division of Site Safety and
Environmental Analysis, NRR

Enclosures:
1. CFR, Title 10
2. NUREG-0472
3. The Reactor Licensing Process