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October 10, 1979

Division of Operating Reactors
Operating Reactors Branch #4
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Robert W. Reid, Branch Chief

Reference: Oregon State University TRIGA Reactor, License No. R-106, Docket No. 50-243

Gentlemen:

In a letter to you dated August 17, 1979 we submitted additional information relating to the proposed amendment of our Technical Specifications. Mr. Vissing, of your office, called us recently and asked that we clarify our answer to question number 10 in our letter of August 17, 1979. He also asked us to comment on the possibility that a water radioactivity monitor be made a Technical Specification requirement for our reactor. We will now address these two issues.

1. Let us first consider question number 10, as submitted to us in your letter of July 20, 1979, and answered in our letter of August 17, 1979. This question postulates a single detector failure and asks whether our new instrumentation system would have an increased or decreased safety margin compared to our present instrumentation system.

   Consider first this failure applied to our new instrumentation. The detector that fails is assumed to be the fission chamber driving the wide-range linear and wide-range log channels. The wide-range linear and log channel, the period circuit, and the period scam would not be functioning as a result of this detector failure. The reactor is assumed to be in the "automatic-servo control" mode--the worst mode for this accident.

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The regulating rod would start driving out as the linear channel signal began to fall, due to the mismatch between the servo demand setting and the linear channel signal. The period-limiting circuit which normally limits the regulating rod speed would not be functioning, since the period signal has been lost. The purpose of this circuit is to limit the regulating rod speed so that the period never gets shorter than about 12-15 seconds. Period scram protection is also lost in this scenario.

Thus, the regulating rod drives out at its maximum speed, the power level increases rapidly, the period becomes quite short (as short as about .15-.20 seconds), and the reactor scrams at 110% of full power (i.e., at 1.1 MW). The scram would be initiated by either the percent power or the safety power channel--two independent channels, each of which have a scram setting at 1.1 MW. The critical parameter--the fuel temperature--would increase about 10°-40°C above the ambient temperature existing prior to the event. The fuel temperature rise will depend on the initial power level (and fuel temperature) existing before the event occurred, hence the 10°-40°C variation.

Now consider this single-detector failure applied to our present instrumentation. The detector that fails now is assumed to be the ion-chamber that drives the linear channel, so that our accident scenario will be as similar as possible to that postulated for the new instrumentation. The reactor is again assumed to be in the "automatic-servo control" mode. With our present instrumentation, however, the ion-chamber driving the log channel and period circuitry is a separate chamber and is still functioning.

The regulating rod would again start driving out, due to the mismatch between the servo demand setting and the falling linear channel signal. The period-limiting circuit is still functioning, however; thus, the regulating rod does not drive out at maximum speed, but at a slower speed such that the period does not become shorter than 12-15 seconds. Therefore, although the period scram is still operable, we would not get a period scram since the scram set point is about 3 seconds.

The reactor power would thus increase until: (a) a high level scram occurred on the percent power channel at 110% of full power (i.e., at 1.1 MW), or (b) the power leveled off at a higher compensated power level. This higher compensated power level would be determined by the amount of reactivity added when the regulating rod moved from its initial position to the fully out position and by the power coefficient of the reactor (the loss of reactivity for a given increase in power).

Our analysis of this accident for our present instrumentation indicates that if the initial power level before the chamber failure is in the range of essentially zero to about 400 kW, a high level scram would not occur. The reactor power would level off at a new, higher, compensated level below 1 MW. If the power level before the failure exceeds about 400 kW, then a high level scram would occur at 1.1 MW since the compensated power level would be above 1.1 MW.

The increase in fuel temperature above the ambient existing prior to the event would be largest if the initial power level were low (in the range of 0-30 kW). In this case, the fuel temperature rise would be about 240°C. For higher initial power levels, the fuel temperature rise would be lower and in the range of 130°- 170°C.

It should be noted at this point that when this accident scenario is applied to either the present or the new instrumentation system, the measured fuel temperature does not approach the LSSS of 510°C. Hence the safety limit for fuel element temperature is not exceeded for either instrumentation system.

In summary, then, this single-detector failure accident produces fuel element temperature rises of 130°-240°C for the present instrumentation system and rises of 10°- 40°C for the new instrumentation system. Thus the margin of safety for the most critical parameter--the fuel temperature--will be <u>increased</u> if we install the new instrumentation system.

2. We would also like to request that you reconsider your proposed modification of our technical specifications regarding a requirement for continuous surveillance of the reactor primary water radioactivity concentration. In support of this request we would like to submit the following information about our existing systems and procedures.

   The radioactivity concentration in the primary water of the TRIGA is normally very low and is often about equal to Corvallis city water background. When radioactivity is detected, it is predominately the short-lived radionuclide sodium-24, which decays with a 15-hour half-life. Therefore, continuous surveillance of the normal levels of radioactivity in the primary water would usually require a very sensitive monitoring system not presently available on the OSU TRIGA reactor. However, we believe that NRC requirements are presently in effect through our Technical Specifications and the Federal Regulations which require monitoring systems capable of detecting increases in primary water radioactivity associated with abnormal or undesirable circumstances (e.g., leaking fuel).

   The TRIGA reactor primary water system, as supplied to OSU by General Atomic (and as described in our Safety Analysis Report dated August, 1968), included a continuous primary water radioactivity monitor. This monitoring device utilizes a metal-walled Geiger-type detector which is inserted into an aluminum clad cavity in an aluminum water tank through which primary reactor water circulates. While our reactor is currently equipped with an operating water monitoring system such as the one described, the system does not possess adequate sensitivity to accurately detect normal operational changes in water radioactivity concentrations in terms of pCi/ml. During our one experience with a leaking fuel element, this system responded to increased radioactivity concentrations in the primary water, but was much less sensitive to the presence of fission product radioactivity than other monitoring systems available to detect such a problem.

   For example, we had no trouble whatsoever immediately detecting the fuel element leak with our reactor top continuous (particulate) air monitor (CAM). This device, which is <u>required by our Technical Specifications to be operating whenever the reactor is operating</u>, collects its air sample <u>directly</u> over the reactor pool and is positioned only five to six feet away to enhance its ability to immediately detect any increases in airborne particulate radioactivity which could originate from the pool.

The CAM has proved in an actual fuel leak at OSU to be extremely sensitive to the expected particulate fission products rubidium and cesium, which are generated by the escape (and subsequent decay) of krypton and xenon fission gases from the reactor pool water. Traditionally, pool-type research reactors have successfully employed such air monitoring devices to indicate leaking fuel, and we have assumed the CAM's importance in our Technical Specifications was based mainly upon this application.

In addition to our present continuous air monitor which responds only to particulate radioactivity, OSU has just purchased and received a new CAM which incorporates both a particulate and gaseous radioactivity channel. This new device will replace the existing single channel system, and we are certain that our capability to detect fuel leaks will be even greater than before with the new system.

Our reactor radiation monitoring program also incorporates other features dealing with the surveillance of reactor water radioactivity. First, we have one channel of our remote area monitoring system located within a few inches of our primary water demineralizer resin tank. The sensitivity of this monitor is such that it routinely shows an increased reading as the resins remove sodium-24 from the reactor cooling water. As a result of this and radiation measurements on the demineralizer tank during our fuel leak, we are certain that this monitor would provide an indication of abnormalities resulting in significant waterborne radioactivity. As a further control measure, we collect and analyze monthly samples of the reactor's primary and secondary water, and a sample of Corvallis city water, which is used to make up primary and secondary water evaporative losses. Analysis includes isotopic identification and a separate tritium determination.

Through this program we have developed an excellent history of the variations to be expected in the primary water radioactivity concentration and the radionuclides which are regularly present. (As noted above, sodium-24 is the primary contributor to the primary water radioactivity.) Furthermore, we perform daily radiation surveys for loose contamination on the reactor top and on equipment used in the reactor pool, which helps to provide a current indication of the water's general radioactivity concentration.

As a result of the NRC required monitoring systems and procedures already in effect at the OSU TRIGA, we believe that we are presently able to detect, in a timely manner, increases in our reactor primary water radioactivity concentration which would be indicative of abnormal or undesirable circumstances. For this reason, we are requesting that no further requirements involving reactor water surveillance be added to our Technical Specifications.

I hope that we have clarified these two issues. If you desire any further information regarding them, please let us know.

Sincerely,

C. H. Wang
Reactor Administrator
Director, Radiation Center

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