Public Service Company of Colorado

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50-267

May 14, 1982
Fort St. Vrain
Unit #1
P-82142

Mr. George Kuzmycz
U. S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20034

SUBJECT: Fort St. Vrain, Unit No. 1
NUREG 0737, ORNL Review

REFERENCE: ORNL Letter Dated
March 16, 1982

Dear Mr. Kuzmycz:

We have reviewed the above referenced ORNL letter and we have the following comments keyed to the numbered paragraphs of the ORNL Report.

Attachment I

1. Comparison with NRC Guidelines

   We have revised Table 4.1-1 concerning notification of State officials of unusual events to indicate that we will notify the State within two (2) hours of an occurrence but in any event within fifteen (15) minute of declaration of an emergency. This revision was made as a result of the NRC Emergency Appraisal Audit and is documented in our response to the Q-2 questions (P-82053).

2. Detailed Recommendations

   We will review the EAL's and attempt to eliminate ambiguity wherever possible.

   Unusual Event

   We will evaluate LOFC's which are less than two (2) hours and establish an appropriate Unusual Event EAL.

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Alert

With reference to Initiating Event 3 it is important to note that this is not only keyed to reactor pressure but also requires the event to be accompanied by area/stack radiation monitor alarms. Normal pressure referenced is based on programmed reactor pressure. Technical Specifications requires the lower limit on reactor pressure to be less than or equal to 50 psig and the resulting action if this limit is exceeded is reactor scram. This action, however, can occur from any number of things including changes in feed water flow, load changes, changes in helium circulator flow, etc. Under these conditions we may well experience a reactor scram on either high or low reactor pressure program without varying the actual reactor pressure at all.

The intent of this EAL was not to necessarily report scrams that may be caused by Technical Specification limits, but rather to develop EAL's that represented a potential hazard to the public.

In this respect the EAL was established at 100 psi accompanied with radiation release. If the reactor scrams at 50 psi, due to LSS settings, the initial corrective action to mitigate the incident has already been taken and no further action should be required. However, if reactor pressure continues to decay and is accompanied by radiation monitor alarms, it is clear that the reactor scram is a result of primary coolant loss and is then beyond the conservative action of the Technical Spoecification and into an emergency situation.

Initiating Event 4 is based on NUREG 0654 Appendix 1 (formerly NUREG 0610). We agree that the 1,000 times normal does not provide the operator with much guidance given the normal levels at Fort St. Vrain. We will re-evaluate this EAL with the attempt to place a radiation level quantification on the EAL. In doing this, however, please recognize that the EAL will no longer be similar to NUREG 0654 and should therefore be evaluated on this basis.

Initiating Events 5 and 6 were intended to mean that loss of power for up to 30 minutes constituted an EAL. If loss of power exceeded 30 minutes then the EAL would be escalated to the next emergency classification. We feel the EAL is straight forward, but we will take another look at it for possible clarification. We believe the EAL requires some common sense. Obviously in terms of ORNL's comment involving 30 seconds, it would take longer than 30 seconds to start and load the diesel generators. Given the fifteen (15) minutes time notification requirement for the State, however, we believe that the EAL as written could be evaluated and appropriately reported.

For Initiating Event 7, we recognize the 90 minute time limit associated with fire water cool down, and this is incorporated in Emergency Procedures. ORNL's comment automatically assumes that any LOFC will be accompanied by fire water cool down which is not the case. Fire water cool down is an extension of the LOFC incident which is handled by our Emergency Procedures and is therefore not addressed as a specific EAL, but is most certainly addressed in the Emergency Procedures, which in turn reference RERP action. Initiating Event 7 is an event intended to be related to a high potential for fuel damage, similar to Item #9 on Page 1-9 of NUREG 0654.

With reference to Initiating Event 8, we will evaluate this further with the attempt to quantify a time frame.

For EAL Number 1, the values are those limits specified by Technical Specifications. We are modifying this EAL somewhat to reflect comments from the NRC emergency appriasal audit.

We do not know what ORNL's comment means concerning "PSC's Action List" regarding a reactor scram or a rapid shutdown. Further clarification is required.

Site Emergency

We agree with ORNL's comment that our release potential does not approach the limits set forth for a Site Emergency for those cases cited. In fact we go through a ficticious exercise each time we plan our annual drill to manufacture source terms that will get us to site emergency levels. The events listed were included primarily to address typical events of NUREG 0610 without regard to potential release. We would be most happy to key all of our events to primary coolant activity and re-evaluate the EAL Tables on this basis. However, it must be noted that the NRC in NUREG 0654, has dropped completely the referenced release potentials from their Class Description section.

General Emergency

As we discussed in the past we do not have a credible accident analysis that would get us to the level of "General Emergency" as defined by NUREG 0610. In this respect Table 4.1-4 was developed only to indicate that the classification called "General Emergency" exists. We are continuing some further evaluation in this area in terms of our inability to depressurize. When we finish these evaluations we will reconsider Table 4.1-4.

Attachment II

In general, the accident analysis does not appear to be in keeping with our previous discussions and does not appear to be in keeping with current industry practices for investigating accidents beyond the design basis.

We felt from our previous discussions that accidents beyond the design basis would be conducted utilizing realistic information with many of the conservative assumptions eliminated. The analysis presented by ORNL appears to be a mix of conservative assumptions on one hand, coupled with more realistic assumptions on the other. We were under the impression from our previous meetings that an important starting point of any accident analysis beyond the design basis was development of realistic source terms based on updated fuel failure models. Instead, the analysis is based on existing source term data. The PCRV depressurization is assumed to occur non-mechanistically at the time of highest circulating activity from fully pressurized conditions, and then utilizes best estimate meteorology for dose calculations. We, of course, have no problem with use of best estimate meteorology data as a realistic approach to the accident, but the source term also needs to be realistic.

We agree that an elevated release and shifting wind patterns will have an effect on dose calculations. We are not familiar with the PUFF code. We are very familiar with the CRAC code which is the presently accepted state-of-the-art for calculating reactor accident consequences. We are not in a position, therefore, to make any comments concerning the PUFF code or its continued development.

Although the method of analysis is perhaps not totally representative it is nice to note that even with some conservative assumptions an extension of the design basis accident to a more severe sequence still results in whole body exposures a factor of 100 below the whole body PAG and a factor of two (2) below the lower PAG for thyroid exposure.

We note that ORNL is conducting further studies. Perhaps these studies could result in more benefit to all parties concerned if they took advantage of other studies involving HTGR's such as the NRC funded HTGR siting study presently contracted to the Idaho National Engineering Laboratory (INEL).

We, of course, are still working through General Atomic Company on accident studies beyond the design basis (alternate LOFC). While this work will not be completed until later this year, the work should serve to offer an effective alternate to the ORNL work.

Attachment III

We have no specific comment.

We have the ORNL reports dated February 16, 1982, and April 16, 1982, under review. We will be forwarding our comments on these reports in the near future.

Very truly yours,

Don W. Warembourg

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