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May 4, 1983

Director of Nuclear Reactor Regulation
Attention: Mr. R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Calvert Cliffs Nuclear Power Plant
Units Nos. 1 & 2; Dockets Nos. 50-317 and 50-318
Main Steam Line Break Inside Containment

References:

(a) NRC I&E Bulletin 80-04, Main Steam Line Break With Continued Feedwater Addition, dated February 8, 1980.

(b) NRC Letter from R. A. Clark to A. E. Lundvall, dated January 20, 1982.

(c) BG&E Letter from A. E. Lundvall to R. A. Clark, dated November 16, 1982.

(d) BG&E Letter from A. E. Lundvall to R. A. Clark, Sixth Cycle License Application, dated February 17, 1982.

Gentlemen:

On April 30, 1983 BG&E contacted the NRC and reported that a deficiency had been identified in the assumptions used to evaluate the potential effects of a main steam line break inside containment. This letter describes the circumstances that resulted in this licensee event report and constitutes our justification for continued operation.

1. **Description of Analyses**

In response to the concerns raised by I&E Bulletin 80-04 (see References (a), (b) and (c)) BG&E recently performed several main steam line break (MSLB) analyses to evaluate the effects of a failed-open main feedwater regulating valve (MFRV) on peak containment pressure and core reactivity response. The purpose of these analyses was to identify the potential for exceeding containment design pressure or experiencing a return-to-power event, and to provide data that could be used to support any corrective actions that might be deemed necessary.

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The results of these engineering-oriented analyses indicated that with the current feedwater system design, the consequences of a MSLB would be significantly worsened by the assumption of a failed-open MFRV.

On April 29, 1983 the Off-Site Safety Review Committee (OSSRC) reviewed these analytical results. The OSSRC determined that although a failure of the MFRV was not considered in the as-licensed design basis for Calvert Cliffs, an appropriate treatment of this non-safety grade component would have been to disallow any credit for its function. On the basis of this determination, the OSSRC concluded that the main steam line break analysis contained in the FSAR was erroneous and that this issue constituted an unreviewed safety question.

On April 30, 1983 a licensee event report was initiated pursuant to paragraph 6.9.1.8.h of the Technical Specifications to inform the NRC of our conclusions. Efforts were immediately begun to quanitify the impact of a MFRV failure on the core reactivity and containment pressure responses to a MSLB for the existing plant configuration. This information was required to support any decision with regard to the safety of continued operations.

An additional MSLB analysis was performed using FSAR methodology to determine the maximum peak containment pressure that would result from runout main feedwater flow to the affected steam generator. This case assumes that a full-size MSLB (guilliotine rupture) occurs during full power operations. Other assumptions used in this analysis include:

a. The reactor coolant pumps are not manually tripped upon SIAS as required by the operating procedures;

b. Only half of the containment cooling and containment spray system capacity is available; and

c. The steam release contains 20% moisture carryover.

This analysis yielded a peak containment pressure of approximately 80 psig and a peak temperature of 306°F.

The analysis that was performed to bound the core reactivity response used a methodology similar to that described in Reference (d) with the exception that a 1300gpm auxiliary feedwater flow was assumed to initiate at 180 seconds (and was not isolated), a six-second MSIV closure time was assumed, and a 60-second MFIV closure time was used. Two hot-full-power cases were examined, one assuming a single stuck-out CEA and one assuming that all CEAs scram and both safety injection trains operate.

For the case with the stuck CEA, negative reactivity credit was assumed during return-to-power due to the local heating of the inlet fluid in the hot channel which occurs near the stuck CEA. This credit is based on three dimensional coupled neutronic thermal-hydraulic calculations performed with the HERMITE/PORC Code. As a result of the continued excess auxiliary feedwater flow, this analysis resulted in a peak return-to-power of about 10% at 400 seconds and would show some fuel failures.

For the case where all CEAs scram, the resulting core reactivity is about $-0.5\% \Delta\rho$. There is no return to power and no fuel failures.

2. **Discussion of Conservatisms**

A significant amount of conservatism is inherent to the results discussed above. The major sources of this conservatism are summarized below:

Containment Pressure Response

a. The reactor coolant pumps are assumed to continue running throughout the event. A more realistic assumption would be to assume that the pumps are manually tripped on SIAS in accordance with the operating procedures. Continued operation of the pumps results in higher heat transfer in the steam generators and consequently results in a higher peak containment pressure than if the pumps were tripped.

b. The analysis only assumed credit for half the containment cooling capacity (coolers and sprays).

c. A delay time of 60 seconds was assumed for the delivery of water to the containment spray header. A more realistic assumption would be a delay time of 30 seconds. Earlier spray delivery would reduce peak containment pressure.

d. The main feedwater isolation valve was assumed to close in 80 seconds upon receipt of a steam generator isolation signal. Experience with surveillance testing of this valve indicates that it will close in 60 seconds, thus reducing the total amount of feedwater introduced to the affected steam generator.

e. Feedwater flow was assumed to continue at runout conditions during the period when the MFIV was closing. A more realistic treatment of feedwater flows would be to evaluate the throttling effect of this valve as it closes. Consideration of this effect would decrease the total flow to the affected steam generator and would yield a lower peak containment pressure.

Core Reactivity Response

a. Item (e) above also applies to the core reactivity response in that any reduction in the amount of feedwater delivered to the affected steam generator will reduce the magnitude of RCS cooldown;

b. A 1300 gpm auxiliary feedwater flow (AFW) far exceeds the current setpoint;

c. The analyses assume a conservative end-of-cycle moderator temperature coefficent;

d. Partial failure of safety injection is assumed;

e. No credit is taken for concentrated boric acid addition from the charging pumps after SIAS;

f. The analyses assumed a conservatively low value for boron reactivity worth ($-1.0\%\Delta\rho$ per 95 ppm);

g. Auxiliary feedwater temperature was assumed to be $40^{o}F$;

h. No mixing in the reactor vessel inlet plenum is credited (the coldest cold leg temperature is used);

i. The cool-down assumes a $6.3\ ft^2$ pure steam break;

j. The steam generator is assumed to blow down to atmospheric pressure;

k. Primary-to-secondary heat transfer is not reduced as steam generator level decreases;

l. No operator action is credited to secure AFW flow;

m. Less HERMITE credit was taken than is expected to be justifiable; etc.

3. **Discussion of Potential Consequences**

We have reviewed the potential consequences of a MSLB break inside containment and have determined that the offsite doses that could result from a failed-open MFRV would be within a small fraction of that allowed by 10 CFR Part 100 for design basis events.

A return-to-power in the core would only occur if the most reactive CEA stuck in the withdrawn position. The return-to-power transient would have to be of sufficient magnitude to cause departure from nucleate boiling before fuel failures would begin to occur. However, such failures will be localized to the vicinity of the stuck-out CEA. Any fission products released from failed fuel pins would be contained within the reactor coolant system. Even assuming that a leakage pathway existed to the containment through the steam generators (as the result of increased steam generator tube leakages) releases to the environment would be insignificant unless a leakage pathway had already been established through the containment. Other minor release pathways may exist through auxiliary systems (e.g. via the auxiliary feedwater turbine steam exhaust) or through the secondary system if it were assumed that the tube leakage occured in the intact steam generator.

The analyses that have been completed to date clearly indicate that the assumptions that yield an unfavorable core reactivity response tend to ameliorate the containment pressure response (and vice-versa). For example, in order to achieve a return-to-power condition in the core, it is necessary to assume that the reactor operator trips the reactor coolant pumps upon SIAS actuation in accordance with the emergency procedures or that the pumps are tripped as a result of a loss of off-site power. This assumption results in the most severe cold-edge temperatures following a MSLB. In the case of the containment analysis; however a trip of the reactor coolant pumps upon SIAS reduces the peak pressure by several up to 10 psig as a result of the reduced heat transfer in the affected steam generator.

As previously noted, the peak containment temperature was calculated to be 306°F. Although this exceeds the existing design limit of 276°F, the duration of the event is too short to adversely affect the operation of safety-related equipment located inside containment or to degrade containment structural integrity.

The containments for Calvert Cliffs were designed to an internal accident pressure of 50 psig. The containments were tested to 57.5 psig during structural integrity tests. The containments were further evaluated for a load combination which includes a 1.5 load factor on the design pressure. Therefore, it can be concluded that the containments were adequate for an internal pressure of 75 psig with sufficient margin as required by the code.

To further study the capacity of the containments beyond the design conditions, it is evident that additional margins are provided in the code allowables and the actual material strengths. Although actual material strengths have not been tabulated accurately for Calvert Cliffs, it is reasonable to assume at least 20% higher material strengths were provided based on past records and experience. Together with a margin of 10% provided in the code allowable, a total of 30% margin over the pressure of 75 psig can be expected. This amounts to a pressure of 97.5 psig. The highest stressed section under this condition is at the cylinder-base junction.

Studies indicate that with regard to containment structural integrity at elevated pressures, electrical penetrations are the limiting components. The Ampheriol canister electrical penetration used at Calvert Cliffs has been tested at 62 psig. Conax penetrations have been tested for as high as 100 psig.

To bound all possible scenarios, we have considered the potential consequences that would be associated with the following concurrent events:

a. Localized fuel failures due to a return-to-power tranisent,

b. Primary-to-secondary leakage as the result of increased steam generator tube leakage, and

c. Loss of containment integrity due to a failed penetration.

The radiological source term for this scenario would be limited by the fact that fuel failures in the core would be restricted to the vicinity of the stuck-out CEA. This source term would be less than that assumed for the maximum hypothetical event. Transport of these radionuclides to the containment would be dependent upon the size of the primary-to-secondary leak and would be hindered by the absence of forced reactor coolant flow (RCPs tripped upon SIAS). Consequently, it is not likely that any but a small fraction of the total radioactivity released from the fuel would be introduced into the containment. Of this, it is expected that only a small fraction would ultimately be released to the environment. This is due to the fact that, unlike in the case of a LOCA, elevated pressures in the containment would exist for a relatively short period of time. By the time any appreciable amounts of radioactivity had been introduced into the containment, little driving force would exist to maintain leakage across the failed penetration.

Consequently, the maximum offsite doses associated with the scenario described above would be well below 10 CFR Part 100 limits and are bounded by the accident analyses described in the FSAR.

4. **Evaluation of Probability of Occurrence**

To evaluate the likelihood of the event sequence in question, we reviewed the probabalistic data base that has been assembled to support the Interim Reiability Evaluation Program (IREP). In that data base (adapted from EGG-EA-5887), the probability of a pipe rupture for pipes larger than three inches in diameter is identified as 1E-10 per hr per pipe section. If we conservatively assume 60 sections of main steam pipe inside each containment, this yields an overall probability of about 5E-5 per reactor year of operation. The probability assigned to the failure of an air-operated valve to close upon demand is 3E-4. Thus, the overall probability of occurrence for a major pipe rupture such as a MSLB concurrent with a failed-open valve (such as the MFRV) can be placed at approximately 2E-8 (plus or minus one or two orders of magitude). This is substantially less than the 3E-4 valve that is normally assigned to the probability of a large break loss-of-coolant accident (LOCA).

It should be noted that the overall probability discussed above applies only to a scenario that could lead to exceeding the containment design pressure and does not consider the magnitude of the pressure response, nor does it include the likelihood of a subsequent loss of containment integrity.

To assess the overall probability of a sequence of events that could result in a return-to-power in the core, the likelihood of a failure of a single CEA to scram must be included. The IREP data base provides a value of 3E-5 per demand for this failure probability. Thus, the overall probability of a MSLB that results in a rapid RCS cooldown and subsequent return-to-power can be estimated to be on the order of 1E-12 per reactor year of operation. This value does not consider the lower probability that would be associated with a return-to-power event that could lead to fuel failures (violation of DNBR) in the vicinity of the stuck CEA.

5. **Corrective Measures**

Based on the favorable results obtained from the engineering analyses performed in response to Bulletin 80-04 concerns, BG&E is proceeding with feedwater system modifications that will provide at least two barriers to the continued addition of feedwater to the affected steam generator after a MSLB. Two modifications are currently being pursued, either of which should result in acceptable peak containment pressures and core reactivity responses.

The first modification includes an automatic trip of the feedwater system (main feed pumps, heater drain pumps, and condensate booster pumps) on high containment pressure.

The second modification involves decreasing the closure time for the MFIV to about 15 seconds. This modification requires installation of a new valve actuator or possibly replacement of the entire valve.

As an interim measure to mitigate the effects of a MSLB concurrent with a failed-open MFRV, the operating procedures have been modified to require manual trip of the feedwater system upon SGIS. Current information indicates that insufficient time would be available to allow the operator to complete this action in time to prevent exceeding the containment design pressure; however, a trip of the feedwater system within the first sixty seconds would have a mitigating effect on the event.

The operator response time is severely restricted by our conservative treatment of flashing in the feedwater piping after the pumps have been tripped. We are currently reevaluating this phenomenon to determine whether our assumptions with regard to the amount of feedwater that would flash into steam can be relaxed. A reduction in the amount of feedwater flashing assumed after a feedwater system trip would increase the response time available to the operator.

Our schedule for implementation of the feedwater system modification discussed above is as follows:

- Complete MSLB analyses to support engineering validity of both proposed modifications (for submittal to NRC).	May 17, 1983
- Complete licensing grade analyses of MSLB to support each of the proposed (tentative) modifications (for submittal to NRC).	June 1, 1983
- Complete installation of feedwater system trip scheme.	Unit 1 - Prior to start-up after the fall 1983 refueling outage Unit 2 - Nov. 17, 1983 (tentative, subject to equipment delivery)
- Complete MFIV modification	Unit 1 - Spring 1985 refueling outage Unit 2 - Fall 1985 refueling outage (tentative, subject to equipment delivery)

6. **Conclusions**

The POSRC and OSSRC have reviewed the information provided above and have concluded that the continued operation of Calvert Cliffs Units 1 & 2 does not constitute a threat to the health and safety of the public. This conclusion is supported by the following considerations:

a. The analyses that were performed to evaluate the effects of a MSLB concurrent with a failed-open MFRV were conservative;

b. The containment structure should be capable of withstanding peak pressures higher than the maximum calculated value of 80 psig by virtue of the safety margins incorporated into its design. Taking credit for load factors and code allowables, the Calvert Cliffs containment is designed to withstand 97.5 psig. The electrical penetrations have been tested to 62 psig; however, other penetrations of similar design have been successfully tested to pressures as high as 100 psig. Assuming a leak developed in an electrical penetration, the leak would discharge into the electrical penetration room. Air exhausted from this room would pass through particulate and charcoal filters before being discharged to the environment.

c. In the event of a return-to-power transient which ultimately resulted in fuel failures, a pathway would not likely exist for the subsequent release of radionuclides to the environment;

d. The overall probability for a MSLB which could cause fuel failures is very low (about 1E-12 per year); and finally

e. Analytical and engineering work is proceeding on an expedited basis to implement appropriate corrective design measures.

If you should have any questions, please do not hesitate to contact us.

Sincerely,

R. E. Denton

AEL/BSM/pdy

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