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SUBJECT: Forwards info re reanalysis of main steam line break inside containment,per NRC Bulletin 80-004.Reanalysis of containment response determined that containment design pressure exceeded w/o operator action to isolate feedwater.

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**DUKE POWER**

May 27, 1993

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269,-270,-287
Reanalysis of Main Steam Line Break Inside Containment

By letter dated February 8, 1980 the NRC provided IE Bulletin 80-04, Analysis of a PWR Main Steam Line Break (MSLB) with Continued Feedwater Addition. The Bulletin request included the following actions:

1. Review of the containment pressure response analysis to determine if the potential for containment overpressure for a MSLB inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow.

2. Review of the reactivity increase which results from a MSLB inside or outside containment.

3. If the potential for containment overpressure exists or the reactor-return-to-power worsens, provide a proposed corrective action and schedule for completion of the corrective action. Provide a description of any interim action that will be taken until the proposed corrective action is completed.

4. Within 90 days of the date of the Bulletin, complete the review and evaluation required and provide a written response describing the review and actions taken in response to each item.

Duke Power Company staff have been reanalyzing the FSAR chapter 15 MSLB transient in order to determine the limits that this accident might impose on plans for extended fuel cycle designs and reload design optimization. The impact on the specifications for the planned replacement of main feedwater block valves was also being investigated. Reanalysis of the core response to a MSLB yielded acceptable results. Reanalysis of the containment response identified that containment design pressure is exceeded without operator action to isolate main feedwater. Errors in the original B&W analyses (circa 1970) included the absence of modeling of passive structural metal in the RCS as a heat source. The reanalysis indicates that this is a significant heat source.

The acceptance criteria for the MSLB in the Oconee FSAR requires that the core remain intact for effective core cooling, no loss of primary boundary integrity



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occur, and doses remain within 10 CFR 100 limits. These criteria are met with the reanalyses. The offsite dose analysis presented in the FSAR evaluates the MSLB outside containment, thus no credit is taken for containment integrity. The design basis for the reactor building is that it be capable of withstanding the internal pressure resulting from a LOCA, as defined in FSAR Section 15 with no loss of integrity. These plant design basis criteria are met, however as required by IEB 80-04, action is required to limit containment pressurization.

Further detail on the reanalysis and safety significance are provided in the attachment. Briefly, reliance on the ICS and operator action while long term solutions are evaluated and implemented is acceptable based on the following:

- The offsite dose consequences of the MSLB inside containment are bounded by the current FSAR analysis.
- The most probable scenario for MSLB inside containment limits the reactor building pressure to within design limits.
- The results of a recent analysis indicate the Reactor Buildings are capable of withstanding a mean maximum pressure of 144 psig. Further, it has been determined that no major failure of internal civil structures or components is predicted.
- The equipment required to mitigate the consequences of the MSLB is qualified and would perform its safety function.
- The estimated frequency of 2E-07 per year for a non-core damage sequence with potential for containment failure is not considered to represent a significant risk to the public.

Long term resolution and proposed implementation schedules are currently under review and will be submitted to the NRC as a supplement to the IEB 80-04 response by August 19, 1993. This time frame is commensurate with the original Bulletin request of 90 days.

Very Truly Yours,

Joe M Davis
for J. W. Hampton

xc: L. A. Wiens, Project Manager
ONRR

S. D. Ebneter, Regional Administrator
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P. E. Harmon, Senior Resident Inspector
Oconee Nuclear Station

Oconee Nuclear Station
Reanalysis of Main Steam Line Break Inside Containment

Background:

The steam line break accident is described in Section 15.13 of the FSAR. The worst case overcooling accident is the double-ended rupture of a 34 in. main steam line with offsite power available. The criteria for unit protection and the release of fission products to the environment are as follows:

1. The core will remain intact for effective core cooling, assuming minimum tripped rod worth with a stuck rod.

2. No steam generator tube loss of primary boundary integrity will occur due to the loss of secondary side pressure and resultant temperature gradients.

3. Doses will be within 10 CFR 100 limits.

Note that the FSAR acceptance criteria for MSLB do not include containment design pressure as an acceptance criterion. The original analysis was found to be acceptable by the NRC on the basis that the calculated offsite doses would be within 10 CFR 100 limits. The NRC evaluation of the original MSLB analysis did not differentiate between a rupture occurring inside containment and one occurring outside containment, thus no credit is taken for containment integrity. The design basis for the reactor building is that it be capable of withstanding the internal pressure resulting from a LOCA, as defined in FSAR Section 15 with no loss of integrity. These plant design basis criteria are met, however as required by IEB 80-04, action is required to limit containment pressurization.

The original steam line break containment analysis considered the following two cases:

With Integrated Control System (ICS) and operator action

With ICS and without operator action.

In both cases, it was shown that the return to power was acceptable and the resultant reactor building pressure response was within the design limits of the reactor building. For the case with ICS and operator action, there was no return to power and a 13 psi rise in reactor building pressure was calculated. This assumed that the SG blowdown was terminated in approximately 50 seconds. For the case with ICS and without operator action, the resulting cooling effect returned the reactor to 35 percent of rated power in 65 seconds. Assuming a nominal tripped rod worth of 5.66 percent $\Delta K/K$ with the maximum rod stuck out, the reactor would return to less than 25 percent of rated power in about 350 seconds. If the nominal tripped rod worth is used without consideration for the stuck rod, the reactor would remain subcritical. For the case with ICS and no operator action, the reactor building was calculated to pressurize to 38 psig at 250 seconds, and remain less than 59 psig thereafter.

IE Bulletin 80-04 requirements included review of the steam line break accident to determine if the potential for containment overpressure existed as a result of consideration of runout flow from the Emergency Feedwater System or due to other energy sources, such as continuation of main feedwater or condensate flow. Further if the potential for containment overpressure exists, proposed corrective actions and schedules were to be provided. Duke responded to the Bulletin by letters dated May 7, 1980 and July 23, 1980. These responses were based on the existing containment pressure response analysis. On October 14, 1982 the NRC provided their safety and technical evaluation reports which concluded that the Oconee analysis was acceptable and no further action was required.

Discussion of circumstances which led to this discovery:

DPCo staff have been reanalyzing the FSAR Chapter 15 steam line break transient in order to determine the limits that this accident might impose on plans for extended fuel cycle designs and reload design optimization. The impact of this analysis on the specifications for main feedwater block valves, which are being evaluated for replacement, was also being investigated. The focus of these reanalyses was the core response following a steam line break. Once the core response analyses neared completion, analyses were initiated to evaluate the containment response to a steam line break. These activities began in January of 1993. Preliminary results of these reanalyses identified unexpectedly high containment pressure results relative to the existing FSAR containment pressure results. These results prompted evaluations into the causes of the differences in the results. The reanalyses were thoroughly reviewed in an attempt to identify any errors. The FSAR analyses were reviewed, including discussions with B&W regarding modeling details of these 1970 vintage analyses. The first difference was identified as the absence of modeling the passive structural metal in the primary system as a heat source in the FSAR analysis. The reanalyses indicated that this was a significant heat source. The second difference was that the FSAR analyses including different assumptions regarding main feedwater isolation (which involves Integrated Control System response and/or operator action) were not all carried through to the containment response result. Only the impact on the core response was analyzed for all cases. When the reanalyses carried these cases, such as a case without ICS or operator action through to the containment response result, the results were more severe than the cases which were presented in the FSAR.

Once these preliminary reanalyses were completed and determined to be valid, a meeting was convened at Oconee on March 11, 1993 to evaluate the impact. It was then decided to perform additional analyses with the intent of quantifying the peformance requirements on the operator and on the main feedwater control and block valves. The results of these additional analyses were completed and discussed at a second meeting on May 19, 1993.

Key reanalysis results:

The results of several reanalysis cases are as follows:

With Credit For Automatic Main Feedwater Control

The station is normally operated with main feedwater in automatic control, therefore this is the most probable steam line break scenario. Following the steam line break, the reactor and turbine trip in 2 seconds. The control system throttles back feedwater flow to zero and then reinitiates feedwater to maintain the minimum post-trip steam generator level. Provided that the operator manually isolates main feedwater by closing the main feedwater control valves (20 second stroke time) beginning at 170 seconds, the peak containment pressure will remain below the design pressure of 59 psig. These operator action requirements are in current emergency operating procedures and are required to be committed to memory for all licensed operators. Further, simulator training of randomly selected Operations crews during the past week indicates that they take action well within the assumed time of 170 seconds. The estimated probability of this scenario occurring and the operator failing to respond in time to maintain the containment pressure below 59 psig is 1.4E-7 per reactor year.

Without Credit For Automatic Main Feedwater Control

With main feedwater flow in manual control, an unlikely operating mode, a high flowrate continues after reactor trip until feedwater isolation is manually performed by closing the main feedwater control valves. These valves have a 20 second stroke time. This action must be initiated at 25 - 30 seconds in order to maintain the peak containment pressure below 59 psig. These operator action requirements are in current emergency operating procedures and are required to be committed to memory for all licensed operators. Further, simulator training of randomly selected Operations crews during the past week indicates that they take action within the assumed time. The estimated probability of this scenario occurring and the operator failing to respond in time to maintain the containment pressure below 59 psig is 4.0E-9 per reactor year.

Without Credit For Automatic Main Feedwater Control and Main Feedwater Control Valve Sticks Open

Assuming that the main feedwater control valve sticks open, a high feedwater flowrate will continue after reactor trip. The main feedwater block valve (stroke time 120 seconds) must be relied on for isolating feedwater flow. This additional assumption that the main feedwater control valve sticks open significantly reduces the probability of this scenario, which is 4.0E-10 per reactor year. Even though the probability of this scenario is extremely low, operator response to initiate closure of the main feedwater block valve at 120 seconds will limit the peak containment pressure to approximately 140 psig. These operator action requirements are in current emergency operating procedures and are required to be committed to memory for all licensed operators. Further, simulator training of randomly selected Operations crews curing the past week indicates that they take action well within the assumed time.

With a Loss of Offsite Power

If a loss of offsite power occurs simultaneously with the reactor trip following the steam line break, the Feedwater and Condensate System pumps trip due to the loss of power. The operator must isolate feedwater via the main feedwater control valves within 170 seconds in order to maintain the peak containment pressure below 59 psig. The estimated probability of the operator failing to respond and maintain containment pressure less than 59 psig is 1.4E-10 per reactor year.

Consequences of exceeding containment design pressure:

The Oconee Reactor Buildings were originally evaluated for a design pressure of 59 psig. In the Oconee FSAR, in Table 3.8-5, it can be seen that for load combinations including 150% of the internal design pressure, 88.5 psig, the maximum concrete tension in the cylindrical shell was calculated to be 111. psi, 52% of allowable, and the maximum reinforcing steel tension 29315. psi, 81.4% of allowable. The maximum concrete shear stress for load combinations of this type was calculated to be 64.7 % of the allowable.

Therefore, the implied maximum internal pressure without exceeding the allowable stress in the most critically stressed component (reinforcing steel) is 109 psig. The pressure causing yield of the reinforcing steel is 121. psig.

A separate, more recent analysis has also been performed using the ABAQUS computer code. The results of this analysis indicate the Reactor Buildings are capable of withstanding a mean maximum pressure of 144 psig, with a standard deviation of 1.95 psi. It is this pressure at which initial yielding in the hoop tendons is predicted to occur. This same analysis concluded that failure would likely be non-catastrophic tearing of the liner plate near structural discontinuities, with attendant venting through cracks in the concrete sufficient to arrest the rise of pressure in the containment.

The maximum radial deflection of the Reactor Building cylindrical wall for 144 psig was calculated to be 2.5" near mid-height. It has been determined that no major failure of internal civil structures or components is predicted as a result of this deflection.

Review of impact on environmental qualification of equipment in containment:

The electrical equipment required to mitigate the consequences of a MSLB is environmentally qualified under 10CFR50.49. The accident qualification of the equipment is based on the postulated LOCA profile. The postulated LOCA profile remains at peak temperature for approximately 10 minutes and then slowly ramps down over a period of 10 days. The MSLB vapor temperature profile is a rapid spike lasting approximately 55 seconds. Previous industry analysis has shown that equipment internals do not experience the peak temperatures associate with the MSLB vapor temperature spike. Even though the MSLB temperature may be higher for a short period of time, the equipment internal temperature is higher for each point in time when subjected to a LOCA. Therefore, the equipment required to mitigate the consequences of the MSLB is qualified and would perform its safety function.

Conclusions:

The most probable scenario for MSLB inside containment limits the Reactor Building pressure to within the design limit of 59 psig. The estimated frequency of those steam line break scenarios which exceed the design limit is 2E-07/yr, which is very low. Furthermore, these accident sequences do not involve core damage. The Reactor Buildings are capable of withstanding a mean maximum pressure of 144 psig. No major failure of internal civil structures or components is predicted. The equipment required to mitigate the consequences of the MSLB is environmentally qualified and would perform its safety function. Based on the above, offsite dose consequences of the MSLB inside containment are bounded by the current FSAR analysis and the event is not considered to represent a significant risk to the public.