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October 19, 1983

Docket No. 50-255

Mr. David J. VandeWalle
Nuclear Licensing Administrator
Consumers Power Company
1945 W. Parnall Road
Jackson, Michigan 49201

Dear Mr. VandeWalle:

SUBJECT: REVIEW OF NUREG-0737 ITEM II.K.2.17, VOIDING IN REACTOR COOLANT SYSTEM DURING TRANSIENTS

Palisades Plant

NUREG-0737 Item II.K.2.17 required that licensees analyze the potential for voiding in the reactor coolant system during anticipated transients and provide the results to the staff. To comply with the requirements, Combustion Engineering Owner's Group submitted a report entitled, "Effects of Vessel head Voiding During Transients and Accidents in CE NSSS's", CEN-199; March 1982 which you have endorsed. We have completed our review of this report. A copy of our Safety Evaluation Report is enclosed for your information.

We have concluded that the requirements of NUREG-0737, Item II.K.2.17 have been met. Therefore, this completes our review of Item II.K.2.17 for your facility.

Sincerely,

Original signed by
Dennis M. Crutchfield, Chief
Operating Reactor Branch #5
Division of Licensing

Enclosure:
Safety Evaluation

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PALISADES PLANT

MULTI-PLANT ACTION ITEM F-33

VOIDING IN THE REACTOR COOLANT SYSTEM DURING

ANTICIPATED TRANSIENTS IN COMBUSTION ENGINEERING PLANTS

I. INTRODUCTION

During NRC's review of transients in Babcock and Wilcox (B&W) plants after the TMI-2 accident it was noticed that pressurizer water levels did not always change as expected. It was surmised that steam, which formed and accumulated in the hotter, upper-head region of the reactor vessel during the transients, caused this anomaly by acting as a second pressurizer. There was not enough data to determine how much steam was formed during the transient. Also, the formation of steam in the upper head had not been considered in the accident and transient analyses; so a letter (ref. 1) requesting an evaluation was sent to all B&W plants on January 9, 1980.

On June 11, 1980, a steam bubble formed in the upper head region of a Combustion Engineering plant during a natural circulation cooldown (ref. 2). The issue of steam formation in the reactor coolant system (RCS) was thereafter extended to all pressurized water reactors (ref. 3).

The June 11, 1980 event also caused the generation of another NRC Generic Letter (ref. 4) which asked all PWR licensees about their capabilities for performing natural circulation cooldown. The natural circulation issue, which is now called Multi Plant Action

No. B-66, is being evaluated separately even though there is some overlap with this F-33 task action item.

II. DISCUSSION

To comply with task action item F-33 the Combustion Engineering Owner's Group evaluated the potential for and consequences of voiding in the RCS's of all Combustion Engineering plants. This evaluation is described in reference 5.

For this evaluation explicit nodes for the upper head region of the reactor vessel were put into the models for two computer programs. The LTC program, which is one-dimensional and assumes a single phase in the RCS, was used for the normal operational transients. The CESEC program, which uses a node flow path network to model the RCS, was used to analyze the effect of steam voids in the events in Chapter 15 of the Safety Analysis Report (SAR). (Section 14 of the Palisades SAR.

In general, calculations with these computer programs showed that the ratio of upper head volume to total RCS volume is a direct indicator of the impact steam void formation has upon transient RCS pressure. Since this ratio gets larger with plant size the effect of steam void in the upper reactor vessel heads is greatest in the largest plants. Other plant dependent parameters that were taken into account in the calculations are: safety injection set point, high pressure safety injection pump shut off head, auxiliary feedwater flow, and the capacity of the main steam safety valves.

By using the LTC program it was shown that for normal operational transients, including rapid cooldowns after trips from normal operation, the subcooling margin in the upper region of the reactor vessels is at least 30°F. This minimum occurs shortly after a reactor trip from normal operation. After that at operator controlled cooldown rates of up to 100°F/hour, which is a Technical Specification limit, with reactor coolant pumps running, the minimum subcooling margin was calculated to be 46°F for all operating C.E. plants including San Onofre 2 & 3. C.E. concluded that this is sufficient margin to prevent a void from forming in the upper region of reactor vessels during operational transients in which the reactor coolant pumps continue to run.

A loss of offsite power (LOOP) and consequent trip of the reactor coolant pumps was analyzed where it was also assumed that an atmospheric steam dump valve was inadvertently opened. This was found to be the most limiting anticipated operational occurrence and it is more limiting than just a LOOP by itself. The analysis showed that during the pretrip portion of this occurrence the pressure remains above the saturation pressure in the hot, upper-head region of the reactor vessel; so no steam void is formed and there is no impact on the departure from nucleate boiling (DNB) during this time period. Voids begin to form in the upper head region after the reactor is tripped on the low pressurizer pressure trip setpoint. The void volume increases slowly with cooldown until reaching its maximum value near the time of steam generator dryout. Thereafter the void volume decreases slowly because the RCS pressure increases due to decay heat and the high pressure safety injection flow.

Further CESEC analyses showed that: (1) all of the effects of upper head voiding after a main steamline break (MSLB) are more limiting than for the inadvertent opening of an atmospheric dump valve, and (2) the MSLB is the most limiting overcooling event with steam void in the upper reactor head. Once again the analysis showed that during the pretrip portion of the MSLB the pressure remains above the saturation pressure in the upper head region; so no steam is formed and there is no impact on DNB during this time period. After the reactor trip, void formed in the upper head region and held up the pressure. This higher pressure delayed the safety injection actuation signal and reduced the flow from the high pressure safety injection pumps; so that less boron reached the core prior to steam generator dryout. Consequently there was insufficient boron to keep the reactor shutdown in all CE plants and there was a return to power in the largest plants. However, it was found that during the return to power all Standard Review Plan (SRP) acceptance criteria were satisfied for all CE plants when the reactor vessel upper head region void effects were conservatively modeled. The most void was formed in the largest plants, which in this case are San Onofre 2 & 3, but even in these plants there is a minimum of over 200 cubic feet of water between the steam void and the hot legs.

For the depressurization events, LOCA's, including an inadvertent opening of the PORV, were analyzed according to 10CFR50 Appendix K criteria and have been evaluated separately. For the remaining depressurization events the CESEC analyses showed that the effects of steam voids in the upper head are most limiting after a steam generator tube rupture (SGTR), but the amount of void formed is much less than during a MSLB. The major concern for the SGTR is the primary to secondary leakage and consequently the amount of radioactivity released from the secondary side. Voids form in the upper head region after the reactor trips on low pressurizer pressure and forced RCS flow is lost due to a concurrent loss of offsite power. These voids act as a second pressurizer and hold up the RCS pressure so there is more primary to secondary leakage and hence more radioactivity released. However, the calculated increase in released radioactivity is still within the SRP acceptance criteria.

III. EVALUATION

The CESEC computer program, which was used to analyze the Chapter 15,* depressurization and overcooling events with an explicit upper reactor vessel region, has been checked with experiments and approved by the NRC. The staff finds that: (1) the steam void effect on the DNB ratio is negligible and (2) the calculated minimum 200 cubic feet of water between the steam void and hot legs in these reactor is sufficient to prevent the blocking of flow in the hot legs and that the consequences of this steam void in the upper reactor vessel region are acceptable for all of the non-LOCA, Chapter 15* depressurization and overcooling events.

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* Section 14 of the Palidades SAR

IV. CONCLUSION

The staff concludes that the voids generated in the reactor coolant system of Combustion Engineering plants during any anticipated event are accounted for in present analysis models even though these models and analyses are not described in the FSAR's of the older plants. The staff further concludes that this steam void will not result in unacceptable consequences in any of these CE plants.

Date: October 19, 1983

REFERENCES

1. Reid, R. W. "Concern for Voiding During Transients on B&W Plants", dated January 9, 1980.

2. Check, P. S. "Void Formation in Vessel Head During St. Lucie Natural Circulation Cooldown Event of June 11, 1980, dated August 12, 1980.

3. U.S.N.R.C. "Clarification of TMI Action Plan Requirements"; NUREG-0737; page II.K.2.17-1, dated November, 1980.

4. U.S.N.R.C. "Natural Circulation Cooldown (Generic Letter No. 81-21)", dated May 5, 1981.

5. Nuclear Power Systems Division, Combustion Engineering Incorporated; "Effects of Vessel Head Voiding During Transients and Accidents in C-E NSSS's", CEN-199; March 1982.