MISSISSIPPI POWER & LIGHT COMPANY
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February 22, 1985

NUCLEAR LICENSING & SAFETY DEPARTMENT

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Units 1 and 2
Docket Nos. 50-416 and 50-417
License No. NPF-29
File 0260/0272/0756
Quarterly Status Report - December 31, 1984, "Degraded Core Accident Hydrogen Control Program"
AECM-85/0023

The Grand Gulf Nuclear Station (GGNS), Unit 1 Facility Operating License (License No. NPF-29) requires that Mississippi Power & Light (MP&L) submit to the NRC quarterly reports on the status of the "Degraded Core Accident Hydrogen Control Program." In response to that requirement MP&L is herewith submitting this status report. This report covers the time period since October 1, 1984, through December 31, 1984.

Should you have any questions concerning this report, please contact us.

Yours truly,

L. F. Dale
Director

MJM/GWS/SHH:rw
Attachment

cc: Mr. J. B. Richard (w/a)
Mr. N. S. Reynolds (w/a)
Mr. R. B. McGehee (w/a)
Mr. G. B. Taylor (w/o)

Mr. Richard C. DeYoung, Director (w/a)
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Quarterly Status Report for
Quarter Ending December 31, 1984

"Degraded Core Accident
Hydrogen Control Program"

Grand Gulf Nuclear Station
Units 1 and 2
Docket Nos. 50-416 and 50-417

Mississippi Power & Light Company

Quarterly Status Report - December 31, 1984

"Degraded Core Accident Hydrogen Control Program"

1.0 Introduction

This quarterly status report is submitted to comply with a requirement in the Grand Gulf Nuclear Station, Unit 1 Facility Operating License (License No. NPF-29). This requirement specifies that Mississippi Power & Light (MP&L) should provide quarterly reports outlining the status of the on-going research program to address degraded core hydrogen control requirements. This report covers the fourth calendar quarter of 1984 ending December 31, 1984.

This report includes brief summaries of the submittals made by MP&L during this quarter along with summaries of meetings between the NRC staff and MP&L. MP&L is participating in the Hydrogen Control Owners Group (HCOG) which is conducting generic research and completing generic analyses to resolve the degraded core hydrogen control issue. Since the work completed by HCOG complements MP&L's program to resolve this issue, this report also includes summaries of meetings between the HCOG and the NRC. The summaries of these meetings included in this report do not reflect a formal HCOG position with respect to any issue and represent only the MP&L interpretation of the meetings.

2.0 Summary of MP&L Submittals

AECM-84/0492, November 19, 1984

MP&L letter number, AECM-84/0492, dated November 19, 1984 provided a response to a letter dated September 14, 1984 from your staff requesting MP&L to provide additional information on the use of the CLASIX-3 code. We responded by indicating that a detailed program plan was being developed by HCOG and that following review of the plan, we would indicate the extent of our endorsement by January 1, 1985.

AECM-84/0544, December 31, 1984

MP&L letter number, AECM-84/0544, dated December 31, 1984 indicated that HCOG was proceeding towards resolution of the NRC's requests for additional information on the use of the CLASIX-3 code as detailed in the HCOG Program Plan, and would respond to the requests once the acceptance criteria established in the Program Plan was found acceptable. A commitment was made to submit a schedule for completion of this effort following HCOG-NRC concurrence on the Task 8 acceptance criteria.

3.0 Summary of Meetings

HCOG and NRC meeting on October 3 and 4. 1984

The HCOG met with the NRC on October 3 and 4, 1984, to provide presentations on the use of the BWR Core Heatup Code for predicting hydrogen and steam production rates for recoverable degraded core accidents. The presentations included a review of technical details of the BWR Core Heatup Code including the details of core modeling, a review of code solution schemes and an assessment of the sensitivity of code results to input parameter variations.

The objective for the meeting was to have the NRC staff assess the BWR Core Heatup Code adequacy for predicting hydrogen and steam production rates for recoverable degraded core accidents. The release rates are to be utilized in a 1/4 scale test facility to demonstrate the capability of hydrogen ignition systems and to assess the survivability of vital equipment inside containment due to standing diffusion flames.

The meeting began with a brief discussion of the development of the BWR Core Heatup Code version used by HCOG. The initial version of the BWR Core Heatup Code was developed by S. Levy Inc. for EPRI under the IDCOR program. The code has been substantially modified by HCOG to complete the analyses of hydrogen generation in a BWR/6. Technical efforts in modifying the code have been completed by S. Levy under the direction of EPRI.

Several characteristics of the degraded core scenarios considered with the BWR Core Heatup Code were reviewed because the accident analyzed with the BWR Core Heatup Code affects the adequacy of the code for analysis use. The scenario begins with some initial transient or reactor coolant pressure boundary break which results in a reactor scram and vessel isolation. Makeup water is assumed to be unavailable. The emergency procedure guidelines are followed to maintain steam cooling until the reactor pressure vessel is essentially depressurized with the ADS valves open. Up until this point, steam cooling has been effective in cooling the core and no significant core heatup or hydrogen production has occurred. The BWR Core Heatup Code uses this point as the initial condition. It is assumed that the depressurization of the vessel and steam cooling result in the core being mostly uncovered before significant core heatup begins. The core continues to boil off the remaining liquid inventory until recovery of a reflood system which results in termination of the transient by core injection.

It was stated that one of the early objectives in developing the BWR Core Heatup Code was to assure that the code accurately represented differences between a BWR and a PWR core geometry. The code models actual core and reactor pressure vessel (RPV) geometries, materials and masses, and includes a detailed representation of the core spatial power distribution. Energy and mass are conserved within the control volume defined by the code. Convective heat transfer correlations within fuel bundles are based on tube bundle data.

Thermal radiation modeling in the code is based on the GE methodology used in the CHASTE code and involves rod groups, channel walls, control blades, and the shroud and RPV wall in radiant energy exchange. Additional details of the core modeling used in the BWR Core Heatup Code were discussed including; thermal hydraulic and oxidation modeling regionalization of the core, spatial power distribution, structures outside the core, control blade heat up, and core spray into the upper plenum.

The user input and the solution schemes for the BWR Core Heatup Code were discussed. Several of the input parameters reviewed included; RPV boundary conditions, core geometry, power shape factors, decay energy, temperature limits for fuel melt, zircalloy cladding melt, zircalloy oxidation cutoff and control blade melt temperatures. The solution schemes discussed included types of calculations and methods used for determining the coolant inventory for each bundle and bypass region, heat balance, oxidation of surfaces, fuel nodalization and fuel temperatures.

Oxidation modeling of the BWR Core Heatup Code was reviewed. The Cathcart-Pawel correlations are used below the phase transition temperature at 1850°K and the Baker-Just correlation is used above the phase transition.

Oxidation is irreversibly stopped in each node when the node reaches the high temperature of 2400°K. This temperature is well above the zircalloy melt temperature of 2170°K. Zircalloy oxidation has been observed to decrease effectively to zero at very high temperatures in the range of 2200-2400°K. Typical tests which have demonstrated this phenomenon included the KFK simulated fuel rod oxidation meltdown experiments which showed termination of oxidation at temperatures between 2200-2300°K. Another test which demonstrated termination of oxidation was the PBF SFD-1 test which showed termination of oxidation in the temperature range of 2300-2400°K. Finally, extrapolation of data obtained by H. Chung of Argonne National Laboratory showed termination of oxidation at temperatures on the order of 2200°K.

The apparent cause of irreversible oxidation termination is mechanistic. The cause is related to liquefaction of the uranium oxide fuel and slumping of the molten mass of zirconium-uranium-oxide. This irreversibly and drastically reduces the effective surface area available for oxidation. Finally, it locally interferes with steam flow which provides additional steam to the region which is oxidizing.

Since the termination of oxidation is believed to be the result of altered geometry, the effect on oxidation is considered to be irreversible. Once a node in the core model reaches 2400°K, oxidation is stopped in that node from that time forward in the transient. Oxidation does not occur in the node even if the node later cools below the oxidation cutoff temperature of 2400°K.

If the zircalloy oxidation is terminated due to slumping of the molten zirconium-uranium-oxide, then other BWR core regions above and below the node whose temperature exceeded the oxidation cutoff would also be affected. Reduced steam flows would reach nodes above the area which had liquified due to disruptions of the channelized flow paths. Reduced steam flow would also reach nodes below the areas which had liquified due to constriction of the flow paths above. Therefore, oxidation in all axial nodes for a fuel channel which had a single node exceed the oxidation cutoff temperature would be substantially reduced or eliminated. These effects are not considered in the BWR Core Heatup Code. Only those nodes in which the cutoff temperature has been exceeded experience reduced oxidation. Nodes above and below these nodes are considered to continue oxidation with an unperturbed flow channel for the steam feeding the oxidation.

It was noted that the zircalloy oxidation cutoff is modeled in the code as a progressive effect starting at 50°K below the cutoff temperature and decreasing to zero at the cutoff temperature. A cosine function is used to define the decrease in oxidation. This change was implemented to assure stability in the termination of oxidation and resulting calculation of temperatures in the fuel.

For oxidation in the fuel bypass region, both zircalloy channel wall outer surfaces and the stainless steel control blade cladding can oxidize if steam is present in the bypass regions. Steam can be formed in the bypass region as a result of reflood caused by quenching of control blades and channel walls.

It was noted that the BWR Core Heatup Code has an internal energy balance audit subroutine with a built in mass balance. The energy balance in the code demonstrated that the code accurately treats energy exchange within a few tenths of a percent during vessel boildown and reflood through most of the hydrogen production. The energy balance is inaccurate to a few percent late in the hydrogen production transient.

The sample output from a BWR Core Heatup Code run was presented. The output was from a run with a 300 gallons per minute (GPM) with the reflood initiated at 2600 seconds after scram. The core was assumed to be 3/4 uncovered when core heatup began.

The first example of code output was a plot of hydrogen generation rate as a function of time, with core heatup beginning 2000 seconds after the scram. This is based on a study of hydrogen generation events completed for HCOG by General Electric. This assumption is also consistent with the MARCH analysis which was completed by the NRC for Grand Gulf Nuclear Station. The actual time when core heatup begins is not significant since the core decay energy only influences timing of the start of hydrogen production. Once

hydrogen production begins, the energy added to the core by oxidation dominates energy addition. Decay energy does not significantly affect the total amount of hydrogen which is produced by an accident.

The next sample code output presented was a plot of cumulative hydrogen generated during the accident as a function of time. The fraction of the core zircalloy inventory which had been oxidized to produce this amount of hydrogen corresponded to oxidizing 19% of the active fuel cladding.

Three other examples of code output were presented. A plot of the fraction of the active fuel cladding oxidized as a function of time was presented. Another plot showed the fraction of the entire core zircalloy inventory which had exceeded its melting temperature as a function of time. The final example of code output was a plot of steam flow exiting the vessel and water inflow to the vessel as functions of time.

Conservative assumptions incorporated in the modeling of the core with the BWR Core Heatup Code and in calculating hydrogen production were reviewed. Several of the significant conservatisms in the code are discussed below.

An ideal, undeformed core geometry is maintained throughout the transient even if portions of the core exceed melt temperatures.

A high oxidation cutoff condition using temperatures well above the melt temperatures for the respective metals is used to terminate oxidation.

There is no decrease in the oxidation rates due to the presence of hydrogen. This is because a high fuel-cladding gap heat transfer rate is used, the gap is assumed to be filled with pure Helium which has approximately 100 times the conductivity of Xenon or Krypton which eventually poison the fuel-cladding gap gas.

In the analytical approach, a cosine shaped zircaloy cutoff curve is used. This tends to decelerate the temperature rise rates of the fuel as the fuel and cladding approach the oxidation cutoff. This approach assures that some nodes will not reach the oxidation cutoff. Therefore, these nodes will continue to oxidize and produce hydrogen as the core is recovered.

The irreversible termination of zircalloy oxidation at 2400°K postpones any effect from zircalloy liquefaction or slumping until the fuel rods are well above the temperatures at which zircalloy melting occurs is a conservative assumption.

Another conservativism is that the core heatup is started from mostly uncovered conditions which would correspond to the conditions following use of the ADS. This approach prevents accumulation of a thick buildup of retarding zircalloy oxidation on the cladding surfaces which would make them less susceptible to accelerated oxidation.

Two other conservatisms used in the analysis were also noted. First, the analysis is based on assuming that 100% of the decay energy remains in the fuel. In actuality, a significant amount of the decay energy would be released to the reactor coolant through the release of fission products. Second, the hydrogen release histories which will be input to the 1/4 scale test facility will be based on the assumption that hydrogen is released virtually simultaneously from the reactor pressure vessel to the containment. In actuality, there would be some smoothing effect due to the presence of flow obstructions such as the steam separators and dryers.

HCOG concluded their presentations with a discussion on the effects of code input variations on the code predictions. A sensitivity study had been completed to verify that variations in code input or modeling assumptions used would produce reasonable variations in code output. Several system parameters were evaluated in the study. Initial system pressure was varied between two and forty atmospheres. The timing for the start of uncovering the core was varied between 2000 and 7200 seconds. The initial core water level was varied from completely covered to being initially three quarters uncovered. The reflood injection rates were varied between 80 GPM and 5000 GPM. Finally, a second boiloff cycle was analyzed for one BWR Core Heatup Code run.

The sensitivity study also included evaluation of several code options and verification of modeling assumptions. The number of unit cells was varied between four and eight. Two options on thermal radiation were evaluated. The first option neglected radiation between unit cells and assumed that each unit cell was isolated from radiant energy exchange. The second option modeled the channel walls and control blades with four nodes and included calculation of radiant energy exchange between unit cells. The zircaloy oxidation cutoff temperature was varied between 2173 and 2400°K. The fuel cladding gap heat transfer modeling was varied. Finally, the fuel rod modeling was modified to incorporate models used by Oak Ridge National Laboratory.

The NRC caucussed following the presentations by HCOG. When the meeting resumed, the NRC staff and their consultants commented on their initial review of the BWR Core Heatup Code. Several questions on the code and its sensitivity to various assumptions were identified.

These questions resulted in discussions on the following areas; fuel cladding gap conductance, initial water level in the core, steam in the core bypass region, modeling of the vessel upper plenum, temperature oxidation cutoff, and loss of control rods due to high temperature in the bypass region. The most significant of these issues appeared to be the use of an oxidation cutoff temperature to terminate oxidation.

HCOG concluded the meeting with a brief summary of the group's use of the BWR Core Heatup Code. HCOG stated that they had started using the code to calculate high hydrogen release rates for use in the 1/4 scale test program for defining diffusion flame thermal environments. Numerous conservative assumptions had been made to maximize the hydrogen production predicted by the code. HCOG believes that this has resulted in a conservative calculation of the amount of hydrogen which could be produced by a degraded core accident.

4.0 Test Program Status

The summaries and status of the HCOG test program as stated here do not reflect the HCOG position with respect to any test program and represent only an MP&L interpretation of these programs.

4.1 1/4 Scale Test Program Status

All major construction tasks are complete, including installation of all instrumentation. Shakedown testing required for verifying proper operation of facility systems and instrumentation is in progress.

4.1.1 Planned Activities for the 1st Quarter of 1985

Complete shakedown tests to verify proper operation of facility systems and instrumentation. Begin scoping tests to assess the effects of variations in key parameters.