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January 31, 1986

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: 0260/0756/7-390.0
Response to NRC Request for
Additional Information
AECM-86/0009

REFERENCES: 1. Letter from W. R. Butler to O. D. Kingsley, Jr. dated January 17, 1986 (MAEC-86/0022)

2. Letter from O. D. Kingsley, Jr. to H. R. Denton dated December 27, 1985 (AECM-85/0367)

On December 27, 1985, in a letter from O. D. Kingsley, Jr. to H. R. Denton, Mississippi Power & Light (MP&L) requested an amendment to License NPF-29 for Grand Gulf Nuclear Station Unit 1. This amendment proposed modification of the License Condition requiring resolution of the hydrogen control issue prior to restart followin the first refueling outage by requiring, instead, resolution on a schedule which reflects the requirements of the recently published Final Hydrogen Control Rule. It was also proposed that if following completion of the MP&L program on hydrogen control it was determined that plant modifications were required to obtain final NRC approval that an adequate hydrogen control system for the plant was installed, then the modifications would be completed on a schedule which was approved by the NRC. The MP&L hydrogen initial program is based on the Hydrogen Control Owners Group (HCOG) Program and the schedules for the MP&L plant specific work and the HCOG generic work are closely related.

On January 17, 1986, in a letter from W. R. Butler to O. D. Kingsley, Jr., the Nuclear Regulatory Commission (NRC) requested that certain additional information be submitted in order for the staff to complete its review of the MP&L application for license amendment and associated proposed schedule for submittal of the final analyses required by the Rule. The attachment to this letter contains the response to your staff's request for additional information as detailed in the January 17, 1986 letter. In order to properly address the

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staff concerns, the questions identified in the January 17, 1986 letter have been paraphrased in the attachment and are addressed individually. This additional information should be sufficient to enable your staff to complete its review as scheduled in order to meet our requested response date of February 28, 1986.

Although not specifically addressed in the Attachment, the staff indicated in Reference 1 that the proposed MP&L schedule for completion and submittal of all elements of the final analysis required by the Final Hydrogen Control Rule is December 31, 1986. This date was established in June 1985 and was based on the HCOG schedule for program completion at that time. The HCOG program is a comprehensive program which has been extensively reviewed by the NRC and is responsive to NRC concerns. Substantial progress has been made by HCOG, but the date of December 31, 1986 is no longer achievable.

On June 24, 1985, MP&L had submitted in a letter from L. F. Dale to H. R. Denton (AECM-85/0182) a proposed schedule for meeting the requirements of the Rule. This schedule was developed based on tasks identified in the HCOG Program Plan and utilized a detailed generic schedule developed by HCOG for identifying schedule milestones. It was believed to be a realistic schedule which provided some contingency for schedule delays but was optimistic in that it was extremely sensitive to any delays in the 1/4 scale test program. As with any major research program it was apparent that inherent uncertainties were included in the schedule. MP&L had indicated upon submitting this schedule that revisions would be necessary to account for unavoidable delays in testing at the 1/4 scale test facility or for any NRC required changes to the hydrogen control program.

Since submitting the proposed schedule in June of 1985, unavoidable delays in the 1/4 scale test program have occurred. Some of the significant delays include: five additional scoping tests necessary to address NRC concerns, an extended power outage at the test facility caused by downed power lines from Hurricane Gloria and, most recently, significant instrumentation modifications required to ensure adequate measurement of peak gas temperatures inside the quarter radius. In addition, a number of facility changes have been made to address NRC concerns, and a portion of the delay is attributable to this. An example of an item that has contributed to the overall schedule delay is the instrumentation modification effort which alone resulted in a two to three month schedule delay for completion of the 1/4 scale test program. These delays were discussed in a meeting with your staff on November 20, 1985. MP&L is currently revising the June 24, 1985 submittal and now expects to show a revised date of approximately June 30, 1987 for completion of the program. This revised date will be for completion and submittal of all elements of the final analysis required by the Rule and does not include NRC review time or plant modifications necessary for final NRC approval.

MP&L therefore proposes that the license condition for completion of the hydrogen control research program be consistent with the schedule for meeting the Hydrogen Rule. MP&L believes that this is necessary because of the magnitude of this hydrogen research test program. The license condition should allow for the inherent uncertainties in the schedule and unavoidable delays in the test program.

Should you have any additional questions or comments please do not hesitate to call.

Yours truly,

L. F. Dale

L. F. Dale
Director

MJM/GWS/SHH:bms
Attachment

cc: Mr. O. D. Kingsley, Jr. (w/a)
Mr. T. H. Cloninger (w/a)
Mr. R. B. McGehee (w/a)
Mr. N. S. Reynolds (w/a)
Mr. H. L. Thomas (w/o)
Mr. R. C. Butcher (w/a)

Mr. James M. Taylor, Director (w/a)
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Dr. J. Nelson Grace, Regional Administrator (w/a)
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NRC Comment 1

Reference 1 stated that MP&L appeared to have excluded degraded core accidents from consideration in their analyses by stating in the significant hazards consideration that the proposed amendment was not related to design basis accidents as previously analyzed in the FSAR.

Response

As indicated on January 6, 1986, in a telephone conversation with your staff, MP&L did not exclude degraded core accidents from consideration in its significant hazards consideration analysis. The statement regarding the absence of degraded core accidents from FSAR accident analyses was intended as an indication of the low probability of a degraded core accident which results in the release of significant quantities of hydrogen. The statement used in the significant hazards consideration analyses can be revised to read, "The amendment does not involve a change to either the hardware, logic or procedures that would result in an effect on the probabilities or consequences of accidents as previously analyzed." As evidenced by the numerous submittals made by MP&L in support of the staff safety evaluation reports and supplements on Grand Gulf (specifically SSER 2, 3, 4 and 5), MP&L has completed and submitted extensive analyses on degraded core accidents and associated sequences involving the generation of significant amounts of hydrogen.

NRC Comment 2

In the significant hazards section of Reference 2, the second sentence of the first paragraph states that "although test results and analyses in Supplement 5 to the GGNS Safety Evaluation Report (SER) suggested that more severe thermal conditions may exist in the containment than previously considered, preliminary results from the 1/4 scale test program indicate that those test results and analyses were overly conservative due to the lack of applicability to the Mark III containment." Please provide additional information to support this conclusion.

Response

The MP&L conclusion that the 1/20 scale test results and analyses were overly conservative due to the lack of applicability to the Mark III containment is consistent with the preliminary NRC conclusions relative to 1/20 scale test data. As noted in the Conclusions section of SSER 5, Section 22, Item II.B.7/8: "...this conservative approach of utilizing unrealistically large hydrogen release rates for a long duration was shown to produce a similarly unrealistic severe local thermal environment."

As reflected in SSER 5, the 1/20 scale program (unlike the 1/4 scale test program) was designed to provide a visual record of hydrogen combustion behavior in a Mark III containment. Although certain limited thermal environment data was obtained, this program was not intended to provide a detailed mapping of the thermal environment resulting from hydrogen combustion in a Mark III containment. In fact, as discussed below, the data obtained was

deliberately taken from the hottest local area in the test facility to provide an upper bound on the thermal environment. The test data was used, however, in preliminary survivability analyses to characterize the potential effects of hydrogen combustion on equipment in the Grand Gulf containment.

In order to better define the thermal environments produced from hydrogen burn events in a Mark III containment, the 1/4 scale test facility was constructed. The design of this test facility, the arrangement of its instrumentation, and the delineation of the test sequence support the goal of clearly defining temperature profiles that will be used in forthcoming equipment survivability analyses.

As a result of the 1/4 scale test program, HCOG has identified several parameters that indicate that the 1/20 scale test data is either excessively conservative or non-representative of a hydrogen combustion event in a Mark III containment. The following summarizes these parameters.

1) As discussed previously, the 1/20 scale test facility was designed to provide a visual record of hydrogen combustion in a Mark III containment. For this reason, temperature data recorded during these tests was limited to a few test locations, and were focussed primarily on the "hot" areas of the facility. The thermal environment for 1/20 scale was defined for the highest temperature region which is directly above the steady diffusion flame and not in areas where essential equipment is located. When preliminary analyses were completed using this data, the result was the application of inappropriately high temperatures to areas that had significantly less hydrogen combustion. The 1/4 scale test facility on the other hand has a comprehensive array of instrumentation whose arrangement has evolved throughout the scoping test program and provides a detailed mapping of the temperature fields throughout the facility.

2) As noted previously and discussed in SSER 5, the hydrogen release histories used in the 1/20 scale test program are excessively conservative and are not representative of the degraded core accident scenarios which may lead to hydrogen generation. The hydrogen release histories used in the 1/4 scale test program are considered to more closely model the degraded core events associated with hydrogen generation. These BWR Core Heatup Code predicted release histories have been reviewed and approved by the NRC.

3) Relatively little thermal gradient data was obtained from the 1/20 scale tests. Although radial and circumferential thermal gradients were not measured in the 1/20 scale they do exist. Thermal gradient data from the 1/4 scale test facility indicates that lower temperatures exist at essential equipment locations, even in the hot chimneys.

4) Because of scaling relationships, the 1/20 scale test data predicts higher flame heights, than would occur at full scale causing a correspondingly "hotter" thermal environment than that which exists at 1/4 scale or that would be expected in a Mark III containment. The 1/4 scale test facility Froude modeling provides more representative flame characteristics.

It should also be noted that oxygen starvation occurred in the wetwell region of the 1/20 scale test facility. This resulted in taller flames that would frequently detach from the pool surface and cause an increase in measured temperatures at the HCU floor elevation. The oxygen starvation was a result of the low design pressure of the 1/20 scale test facility (approximately 5 psig) which required venting the facility during tests to avoid exceeding the design pressure due to temperature increase and corresponding pressure increase associated with the burning of hydrogen. This reduction in air mass in the 1/20 scale test facility resulted in oxygen starvation at the end of some extended duration tests. The 1/4 scale test facility unlike the 1/20 scale was designed as a pressure vessel with a design pressure of approximately 40 psig. The oxygen starvation phenomena was not expected and has not been observed in the 1/4 scale test facility.

5) The 1/20 scale tests did not include simulation of containment sprays which have a significant effect on the global circulation patterns in a Mark III containment. Containment sprays which are conservatively modeled in the 1/4 scale test facility tend to reduce peak and background gas temperatures throughout the test facility.

6) In addition, neither the 1/4 scale or the 1/20 scale tests model the direct local cooling effect of the containment sprays on equipment. This affect would be present in a postulated hydrogen generation event and not including the affect is an additional conservatism that exists in both test programs.

With regard to the ability of the GGNS equipment to survive a hydrogen burn event, scoping test data was reviewed to define the thermal gradients present in the 1/4 scale facility. As indicated in recent meetings between the NRC and HCOG, peak temperatures occur at the 1/8-radius in the wetwell and decrease as vertical distance from the wetwell and radial distance from the drywell wall increases. A review of the Grand Gulf essential equipment list has determined that the essential equipment for Grand Gulf is not located in regions of peak temperature. The equipment has generally been located outboard of the 1/4 radius and above the HCU floor where temperatures are relatively benign. The application of less severe environments than were used in the 1/20 scale equipment survivability analyses (which, as discussed later, showed equipment survival) is expected to confirm the ability of the GGNS equipment to function following a hydrogen burn event. In addition the final GGNS equipment survivability analyses will be further complemented by heat transfer analysis using the HEATING-6 heat transfer code and equipment modeling techniques that have evolved since the completion of the 1/20 scale program.

For these reasons, MP&L believes that the results from the 1/4 scale test program will show that the results from the 1/20 scale test program are overly conservative due to the lack of applicability of the thermal environments to the Mark III containment.

NRC Comment 3

In the third paragraph of the significant hazards section of Reference 2, MP&L states that "the results of the 1/4 scale test program are, at worst, expected to lead to very few minor modifications to further assure equipment survivability." Please provide additional information to support this conclusion.

Response

MP&L has indicated in the response to the previous comment that the measured thermal environment in the 1/20 scale test facility is less applicable to the Mark III containment than the measured thermal environment in the 1/4 scale test facility. The 1/4 scale test data has shown that even though relatively little thermal gradient data was obtained from the 1/20 scale tests, radial and circumferential thermal gradients do exist and tend to lower temperatures at essential equipment locations, even in the hot chimneys.

Previous analyses submitted by MP&L have demonstrated that equipment will survive the hydrogen burns associated with a degraded core accident even using the overly conservative, nonrepresentative thermal environment from the 1/20 scale test facility which was defined for the highest temperature region directly above the steady diffusion flames. When analyses are completed using the more representative thermal environment as expected in the Mark III containment and by evaluating temperature profiles in areas where essential equipment is actually located, it is expected that the results of the 1/4 scale test program will, at worst, lead to very few equipment enhancements to ensure survivability.

Therefore, it has been concluded, as stated in the significant hazards consideration of the December 27, 1985 submittal, that results from the 1/4 scale test program are expected, at worst, to lead to very few minor modifications to further assure equipment survivability.

NRC Comment 4

Address the first standard in 10 CFR 50.92 by supplying additional information for concluding that the proposed change does not involve a significant increase in the probability of an accident previously evaluated. This additional information should address the change in probability for the interval of time from startup following first refueling outage to the date when required modifications, if any, will be completed. This additional information should also supplement the existing analyses for concluding that there is no increase in probability of an accident previously evaluated by assuming that modifications will be made to equipment and emergency procedures based on the outcome from tests and analyses being completed to resolve the hydrogen control issue.

Response

The probability of a degraded core accident which leads to hydrogen generation is very small. This was the basis on which the current GGNS license condition was premised, and this basis has been substantiated by HCOG in its delineation of probable accident scenarios as addressed by documents submitted in support of Task 1 of the HCOG program plan. The proposed amendment to the Grand Gulf Operating License, which changes the required date for obtaining NRC approval of the adequacy of the installed Grand Gulf hydrogen control system from the first refueling outage to a schedule which was developed in accordance with the provisions of 10 CFR 50.44, does not involve a significant increase in the probability of the hydrogen generation accidents previously evaluated. As discussed below, not only does the proposed amendment not involve a significant increase in the probability of occurrence of an accident previously evaluated, but it actually has no effect on that probability.

In the significant hazards consideration portion of our December 27, 1985 submittal, we stated that there was no increase in the probability of an accident previously evaluated because the proposed amendment did not involve a change to either the hardware, logic or procedures. In other words, no changes to the plant or operation of the plant would be required as a result of incorporating this proposed amendment into our existing license. Since the hydrogen ignition system is already installed and is designed only to mitigate the consequences of a recoverable degraded core accident in which significant quantities of hydrogen are released and not to reduce the probability of such an accident and since the proposed change to the operating license involves no change to the plant or its operation, neither the probability of a design basis accident as discussed in the FSAR (which are not related to the design of the HIS) or a degraded core accident is affected.

To further evaluate the effect that the proposed amendment could have on the probability of an accident previously evaluated, it was assumed that one possible outcome of the tests and analyses presently being completed would result in modifications (survivability enhancements) to plant equipment and emergency procedures. If we determine that, as a result of ongoing tests and analyses, modifications are required to ensure equipment survival, then the modifications would still have no effect on the probability of an accident previously evaluated. These modifications or equipment survivability enhancements could only affect the mitigation of a degraded core accident and not the probability of one occurring; thus the probability of an accident previously evaluated would not change for the interval of time from startup following first refueling outage to the date when required modifications, if any, will be completed.

NRC Comment 5

Address the first standard in 10 CFR 50.92 by supplying additional information for concluding that the proposed change does not involve an increase in the consequences of an accident previously evaluated. This additional information should address the change in consequences for the interval of time from startup following first refueling outage to the date when required modifications, if any, will be completed. This additional information should also supplement the existing analyses for concluding that there is no increase in consequences of an accident previously evaluated by assuming that modifications will be made to equipment and emergency procedures based on the outcome from tests and analyses being completed to resolve the hydrogen control issue.

Response

As stated in the December 27, 1985 submittal, MP&L concludes that the proposed change to the operating license does not involve a significant increase in the consequences of an accident previously evaluated, specifically a degraded core accident. By using 1/20 scale test data MP&L has demonstrated that essential equipment required to maintain the core in a safe shutdown condition, to maintain the integrity of the containment pressure boundary, to mitigate the consequences of a degraded core accident and to monitor the course of the accident is likely to survive the high temperatures associated with hydrogen combustion. As discussed in the response to NRC Comment 2 , these thermal environments were demonstrated to be overly conservative and non-representative of a Mark III containment thus providing assurance that the resulting thermal environments from the 1/4 scale test facility will at worst require few minor modifications to further assure equipment survivability.

Quarter scale test data to date has shown that peak temperatures occur inside the 1/4 radius of the containment near the drywell wall. The highest temperatures have been recorded at approximately the 1/8-radius and drop off significantly as distance from the drywell wall increases. Outside the 1/4 radius and towards the inner containment wall, temperatures appear to be benign and pose no threat to equipment survival. Most Grand Gulf essential equipment is located in this area. Upon completion of the equipment survivability analysis program as detailed in the HCOG program plan, it is expected that earlier conclusions drawn from the 1/20 scale test program thermal environments will be confirmed.

In the significant hazards consideration portion of the December 27, 1985 submittal, it was stated that there was no increase in the consequences of an accident previously evaluated because the proposed amendment did not involve a change to either the hardware, logic or procedures. In other words, no changes to the plant or operation of the plant would be required as a result of incorporating this proposed amendment into our existing license. It is therefore appropriate to conclude that there will be no change in the consequences of an accident as a direct result of incorporating the proposed amendment.

To further evaluate the effect that the proposed amendment could have on the consequences of an accident previously evaluated, it was assumed that one possible outcome of the tests and analyses presently being completed would result in modifications (survivability enhancements) to plant equipment and emergency procedures. To address the assumption that modifications would be required to ensure that plant equipment survives a hydrogen burn, we have conservatively assumed that each essential piece of equipment near the HCU floor has the same potential or likelihood of experiencing failure as a result of the high temperatures associated with a hydrogen burn. This is conservative because of the radial and vertical temperature gradients that have been shown to exist in the 1/4 scale test facility. The hydrogen igniters, temperature elements, and pressure and level transmitters are the only essential equipment on the Grand Gulf equipment survivability list located near or below the HCU floor.

If it is assumed that:

1. hydrogen igniter failures occurred, then it is necessary to address the change in consequences of a degraded core accident without igniters following their failure due to the burning of significant amounts of hydrogen. The igniters would have performed their intended function prior to failure. Based on the most probable and most limiting hydrogen generation event accident scenarios, once the igniters ignite the flammable mixture, continuous sustained diffusion flames become anchored at the pool surface and continue to burn the hydrogen as it is released. These flames are sustained even in the absence of a separate ignition source and reduce the buildup of hydrogen in the containment thus eliminating the need for hydrogen igniters to act as separate ignition sources. Therefore, no significant increase in the consequences of a degraded core accident occurs as a result of hydrogen igniter failures following the burning of significant amounts of hydrogen.

2. level and pressure transmitter failure occurred, then it is necessary to address the change in consequences of a degraded core accident without level and/or pressure transmitters following the burning of significant amounts of hydrogen. In the event of such failures, actions specified in the emergency operating procedures would be taken. These procedures are based on the symptom oriented BWR Emergency Procedure Guidelines and provide the operator direction in the event that water level or pressure indication becomes unavailable. The procedure action steps address these events and the execution of the steps have no effect on the consequences of a degraded core accident. Therefore, no significant increase in the consequences of a degraded core accident occurs as a result of level or pressure transmitter failures following the burning of significant amounts of hydrogen.

3. temperature transmitter failure occurred, it is necessary to address the change in consequences of a degraded core accident without temperature transmitters following the burning of significant amounts of hydrogen. In this case, the temperature indication would be off scale high prior to transmitter failure since the thermocouples for the transmitters specified on the Grand Gulf equipment survivability list are located in at least as severe a thermal environment as their associated transmitter. Because of this, there would be high temperature indication prior to failure and emergency operating procedure action steps based on containment temperature would have been taken. Therefore, no significant increase in the consequences of a degraded core accident occurs as a result of temperature transmitter failures following the burning of significant amounts of hydrogen.

NRC Comment 6

Address the second standard in 10 CFR 50.92 by supplying additional information for concluding that the proposed change does not involve an increase in the probability of a new or different kind of accident from any accident previously evaluated. This additional information should address the change in probability for the interval of time from startup following first refueling outage to the date when required modifications, if any, will be completed. This additional information should also supplement the existing analyses for concluding that there is no increase in probability of a new or different kind of accident from any accident previously evaluated by assuming that modifications will be made to equipment and emergency procedures based on the outcome from tests and analyses being completed to resolve the hydrogen control issue.

Response

In the significant hazards consideration portion of the December 27, 1985 submittal, it was stated that there was no increase in the probability of a new or different kind of accident from any accident previously evaluated because the proposed amendment did not involve a change to either the hardware, logic or procedures. As stated in the response to NRC Comment 4, no changes to the plant or operation of the plant would be required as a result of incorporating the proposed amendment into our existing license. Since the hydrogen ignition system is designed only to mitigate the the consequences of a recoverable degraded core accident in which significant quantities of hydrogen are released and not to reduce the probability of accidents, then by not making changes to the plant or operation, the probability of a new or different kind of accident from any accident previously evaluated, including those discussed in the FSAR (which are not related to the design of the HIS) and degraded core accidents cannot be affected.

To further evaluate the effect that the proposed amendment could have on the probability of a new or different kind of accident from any accident previously evaluated, it was assumed that one possible outcome of the tests and analyses presently being completed would result in modifications (survivability enhancements) to plant equipment and emergency procedures. If we determine that as a result of existing tests and analyses modifications will be required to ensure equipment survival, then these modifications will still have no effect on the probability of a new or different kind of accident from any accident previously evaluated. These modifications or equipment survivability enhancements can only effect the mitigation of a degraded core accident and not the probability of accidents occurring. The probability of a new or different kind of accident from any accident previously evaluated would not change for the interval of time from startup following first refueling outage to the date when required modifications, if any, will be completed.