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DUKE POWER

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Catawba Nuclear Station
Dockets 50-413 and 50-414
Reply to Notice of Violation (NOV)
Inspection Report 50-413, 414/96-05

Attached is Duke Power Company's response to the two (2) Level IV violations cited in Inspection Report 50-413, 414/96-05, dated May 31, 1996. These violations were identified during inspections conducted March 24, 1996 through May 4, 1996.

If there are any questions concerning this response, please contact K. E. Nicholson at (803) 831-3237.

Sincerely,

W. R. McCollum, Jr.

\KEN:RESP96.05

xc: S. D. Ebneter, Regional Administrator
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CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
413, 414/96-05-01

Notice of Violation

Technical Specification 3.6.4.3 requires that both trains of the Hydrogen Mitigation System be OPERABLE in MODE 1 and MODE 2 as a Limiting Condition for Operation. Surveillance Requirement 4.6.4.3 states that the Hydrogen Mitigation System is operable provided that 34 of the 35 igniters in each train are operable and inoperable igniters are not on corresponding redundant circuits that provide coverage in the same region of containment. If both igniters on corresponding redundant circuits that provide coverage in the same region of containment are inoperable, actions specified in Technical Specification 3.0.3 are required. Technical Specification 3.0.3, as applied, requires that actions be initiated to place the unit in MODE 3 - HOT STANDBY, within at least 7 hours.

Contrary to the above, both hydrogen igniters in the Pressurizer Relief Tank region of the Unit 2 containment were inoperable from March 18 at 10:40 a.m., until March 19 at 8:58 a.m., for a duration of approximately 22 hours with the Unit in MODE 1.

This is a Severity Level IV violation (Supplement I).

RESPONSE:

1. **Reason for Violation**

Duke Power Company acknowledges this violation. This violation is attributed to an inadequate procedure, IP/2/A/3170/003B, in that it failed to clearly state acceptance criteria involved in surveillance work.

The standard for surveillance type procedures at Catawba Nuclear Station is that acceptance criteria are stated in terms of an expected value (e.g. exact setpoint) and an allowable range (the tolerance). At a minimum where there is no specific tolerance a procedure will state a clearly defined limit which if exceeded will require a specific action. This procedure did not follow this standard in that it did not clearly indicate the acceptance criteria or allowable range. Therefore the cause of this event is an inadequate procedure.

The Licensee Event Report (LER) 414/96-002 submitted for this event also identified inadequate communications as a root cause. While good communications among Maintenance, Operations, and Engineering could have prevented this event, it is our position that a clear acceptance criteria for a surveillance activity will programmatically ensure that surveillance procedures/work orders can only be closed out when a system/component meets its acceptance criteria and that the system/component can only be removed from our Technical Specification action tracking system upon a successful close-out of a surveillance procedure/work order. Therefore, a corrective action to ensure clear acceptance criteria in Maintenance surveilance procedures has been initiated.

Background

On March 8, 1996 a quarterly (92 day) surveillance test, per procedure IP/2/A/3170/003B, was performed on Train B of the Hydrogen Mitigation System. The purpose of the surveillance is to ensure that each group of Train B hydrogen igniters (which are basically glow plugs) and associated indications are operable per Technical Specification 3.6.4.3. Train B has a total of 35 igniters arranged in seven groups. The igniters are checked in discrete groups ranging from two to six igniters.

To perform the surveillance test the technicians are directed by procedure, after taking voltage readings, to calculate the total current flow through each group of igniters. Following the calculations, the technicians perform two conditional procedure steps that direct them to take specific actions if calculated current is less than baseline current or if system is otherwise determined to be inoperable.

The acceptance criterion given in this procedure states that each igniter should draw "~ 1.0" ampere of current. The data sheet on the procedure also gives a "Baseline Current" for each group of igniters. In the case of the group later declared inoperable, the baseline current is given as "5.0" amps, and there are 5 igniters in the group.

The technicians made the required measurements and calculated the current for this group as "4.974" amps. The technicians compared this value to the baseline value of 5.0 amps and noted that it was below the baseline. However, a procedure note indicated that unless the current was more that one amp below required current the circuit was operable. Specifically, this note (Note 2 above step 10.2.15 in procedure) reads, "Tech Spec permits <u>ONLY ONE</u> inoperable igniter per train (ex. measured circuit current is 1.0 or more amps less than required current)." Therefore, the technicians felt they had met the acceptance criteria and had no reason to believe one igniter was inoperable, and certainly no reason to believe the system was inoperable.

The technicians and supervisor did exercise a questioning attitude when the data for all igniter groups was reviewed. They noted that the total current drawn by this group of igniters was lower in relation to baseline than other groups of igniters. But, there was no acceptance criterion for this given in the procedure. The supervisor directed technicians to send a copy of the procedure data sheet to Engineering for additional evaluation per procedural guidance. There was no perceived urgency in this matter since no operational concern was recognized. Given the lack of urgency, the technicians elected to leave a copy of the data sheet at the Engineer's desk with a note attached questioning the data for the igniter group.

Upon completion of the surveillance procedure and having sent the data sheet to Engineering, the supervisor proceeded to close out the surveillance work order. The supervisor concludes that if the engineer wanted to do any further testing based on a review of the data sheet, another work order would be needed. The surveillance work order was taken by supervisor to the WCC (Work Control Center) for sign-off. The WCC SRO (Senior Reactor Operator) confirmed that the surveillance was complete and removed the Train B hydrogen igniters from the TSAIL (Tech Spec Action Item Log).

The engineer reviewed the data sheet from the Train B surveillance test on March 22, 1996. The engineer recognized there was an operability concern with the amount of current being drawn by one of the igniter groups. This recognition was made on the basis of the current reading recorded (4.974 amps) for the igniter group and on trending data. Once the engineer determined that there was an operability concern with a Train B igniter group, the WCC SRO was made aware of the situation. The WCC SRO proceeded to declare Train B Hydrogen Igniters inoperable and make the appropriate TSAIL entry. It was while making the entry into TSAIL that the question arose about the operability of the opposite train of hydrogen igniters. Further investigation revealed that the opposite train had been made inoperable on March 18, 1996 for surveillance testing thereby making both trains of Hydrogen Igniters inoperable which violated TS 3.4.6.3.

2. **Corrective Actions Taken and Results Achieved**

Procedures IP/1(2)/A/3170/03A(B) were revised to clearly define acceptance criteria and to ensure appropriate groups are notified when an out of tolerance situation is encountered.

A change to Technical Specification 3.6.4.3 was submitted and approved by the NRC which adopts the Standard Technical Specification for the Hydrogen Ignition System. This change allows having two igniters inoperable in the same region for up to seven days.

3. **Corrective Action to be Taken to Avoid Future Violations**

Corrective work order 96025349-01 was written to replace hydrogen igniter 2EHM0072. Since a containment entry is required to accomplish this work, this work order will be scheduled during the next Unit 2 outage of sufficient length to complete the work.

A review of Maintenance procedures that implement Technical Specifications surveillance has been under taken to ensure that procedure acceptance criteria/tolerances are clear. This review will be completed by September 1, 1996.

Problem Investigation Process (PIP) 0-C96-0708 was generated and will be the tracking document for completion of this commitment.

4. **Date of Full Compliance**

Duke Power Company is in full compliance.

CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
413, 414/96-05-02

Notice of Violation

10 CFR 50.50, Appendix B, Criterion V, requires that activities affecting quality be prescribed by documented instructions or procedures, and shall be accomplished in accordance with these instructions or procedures.

10.59 requires the performance of an evaluation to determine if changes to the facility (systems, structures, or components) or facility operating procedures described in the Safety Analysis Report (SAR) invovles an USQ safety question.

Duke Power Nuclear Station Directive (NSD) 209, 10 CFR 50.59 Evaluation, Revision 3, effective October 1, 1995, implements the requirements of 10 CR 50.59. Section 209.10.2 of NSD 209 specifies the screening process required to be performed to determine if a facility or procedure change constitutes an USQ safety question which in part requires negative answers to the following questions:

- Does the activity change the facility as described in the SAR?

- Could the activity adversely affect any system, structure, or component that is necessary in accordance with the SAR?

NSD 209 defines the SAR as the set of documents used to support issuance of a plant operating license. These documents include, but are not limited to, the Facility Operating License, the NRC Safety Evaluation Report, the FSAR, the Technical Specifications, and other licensing documents.

Section 101.4.3 of Engineering Directives Manual EDM-101, Engineering Calculations/Analyses, Revision 4, dated March 30, 1995, requires certification of design calculations prior to release of calculation results.

Contrary to the above:

1. The 50.59 evaluation was inadequate in that the negative responses to the NSD 209 questions were incorrect for addressing the February 21, 1996, change to enclosure 4.12 of procedure OP/1/A/6250/02, Auxiliary Feedwater System. Increasing the allowable auxiliary feedwater piping temperature to 250° F changed the design of the auxiliary feedwater system, as described in the SAR. The reduction of the concrete expansion safety factor, from four to two, to permit operability of the auxiliary feedwater piping at a

temperature of 250° F decreased the margin of safety and had a potentially adverse effect on the design of the auxiliary feedwater piping. NRC IE Bulletin 79-02, a licensing document, requires a minimum safety factor of four for concrete expansion anchors.

2. Engineering calculations were released prior to completion of the design certification process, in that on February 21, 1996, a change to Enclosure 4.12 of Procedure OP/1/A/6250/02 was made with uncertified calculations. In changing Procedure OP/1/A/6250/02, for raising the acceptable Auxiliary Feedwater suction temperature, approved February 21, 1996, engineering calculations supporting this change were not approved until on, or after, March 5, 1996. These calculations formed the bases for approval of the procedure change.

This is a Severity Level IV violation (Supplement I).

RESPONSE:

1. **Basis for Disputing the Violation**

Duke Power denies the violation as cited. The reasons for denial of the violation are explained in detail in the following information. In summary, it is felt that the appropriate responses were made in the Unreviewed Safety Question (USQ) screening and that proper safety factors and levels of documentation were provided for a short term operability evaluation.

Background Information

Surveillance commitments require that the auxiliary feedwater pumps be operated on a quarterly basis. Following the running of one of the pumps it was discovered that a check valve downstream of the pump had not seated completely. Backleakage from the valve caused a temperature increase in the pipe on the pump side of the valve. Operating procedures require that the pump be started in order to provide cooling flow to the pipe. Available options to remedy the situation were evaluated by operations and engineering management. Valve replacement was not viable as an immediate option due to long lead times required for design and procurement of nuclear safety related valves. A second option was to evaluate increasing the alarm setpoint temperature at which pump operation was required.

It was determined that engineering would examine the technical basis for the current alarm temperature, and evaluate the possibility of raising that setpoint to reduce or eliminate the need to operate the pumps. Procedure OP/1/A/6250/02, Auxiliary Feedwater System governs the operation of the auxiliary feedwater pumps to cool the system piping when the temperature exceeds the alarm setpoint. Operating the auxiliary feedwater pumps at an increased frequency is generally acknowledged to be undesirable for the following reasons:

a) Operating the pump requires an entry into a Technical Specification Action Statement due to pump inoperability. The Action Statement is entered each time an auxiliary feedwater pump is started in order to cool the piping, due to the system isolations required.

b) The Secondary Thermal Power Best Estimate calculation is affected in a non-conservative manner as it indicates lower than actual Reactor Power on Unit 2 with any auxiliary feedwater pump feeding a steam generator. (Unit 1 calculation inputs are not affected due to different flows.)

c) Operation of the pump represents an operator work around, and leads to operator distractions.

d) Starting the pump every shift constitutes a frequent challenge to an ESF component.

e) Operation of the auxiliary feedwater pumps introduces water that is significantly colder than final feedwater. This requires a reduction in load in order to avoid an increase in Actual Reactor Power (this is applicable to both units at Catawba).

Considering the above, it was determined that long term operation of the auxiliary feedwater pumps on a frequent basis would not be a desirable condition. Continuous pump operation was not an option since the pump turbine steam supply piping is classified as moderate energy piping that cannot be pressurized as would be experienced by continuous pump operation. A modification for the redesign and replacement of the affected check valves was investigated, but was considered a long term resolution due to the long lead time required for the manufacture of nuclear safety related valves.

Under these circumstances, it was clear that a more immediate resolution was required to preclude the frequent pump operation. After consideration of the above factors, management determined that the most conservative alternative from any of the limited options available was to evaluate the effects of increasing the alarm setpoint, thereby decreasing the need to operate the pump. It was determined that increasing the setpoint from 225° to 250° F would significantly decrease the need for pump operation to cool the pipe.

Revising the operating procedure to increase the alarm setpoint required that a 10 CFR 50.59 evaluation be performed per NSD 209. The directive requires that five screening questions be addressed to evaluate applicability of an USQ. In order to adequately answer these questions a technical review of the thermal effects on the pipe itself had to be investigated. This investigation was based upon meeting acceptance criteria for pipe stresses, nozzle loads, clearances, and pipe support qualification. The purpose of this evaluation was to provide assurance that the design and function of the auxiliary feedwater system as described in the SAR would not be changed.

Engineering personnel performed the evaluation by analyzing the affected pipe at the temperature of 250° F. All pipe stresses, nozzle loads, and clearances were determined to remain within normal design limits. Review of the pipe support loads indicated a number of supports with significant load increases. These were evaluated and determined to be within load limits except for a small number of supports with concrete anchor bolt load increases. Normal design criteria per NRC IE Bulletin 79-02 requires a safety factor of four for concrete sleeve anchors; however, for short term operability (until the next refueling outage), a safety factor of two is allowed. Comparison calculations verified that a minimum safety factor of two was met for all the affected supports. The review concluded that less than 10 of the nearly 500 auxiliary feedwater system supports would require permanent physical modifications to meet the safety factor of four.

Upon completion of the engineering evaluation of the pipe and supports, Procedure OP/1/A/6250/02 was revised to allow the setpoint temperature to be increased to 250° F. Station modification plans were initiated and approved to perform the necessary design work required to implement permanent design changes at the next refueling outage. These modifications will upgrade all affected supports to a safety factor of four.

Response to Finding 1

Finding 1 states the NRC's disagreement with the negative response to two of the screening questions used to evaluate applicability of an USQ. NSD 209 stipulates five screening questions for USQ Evaluation, as listed below:

1. **Does the activity change the facility as described in the SAR?**
2. Does the activity change procedures, methods of operation, or alter a test or experiment as described in the SAR?
3. Does the activity appear significant enough to require inclusion in the SAR?

4. **Could the activity adversely affect any system, structure, or component necessary to operate the plant in accordance with the SAR?**
5. Does the activity perform a test or experiment that is NOT described in the SAR?

The violation states that the negative responses to Questions 1 and 4 were incorrect, in that increasing the alarm setpoint of the auxiliary feedwater system piping changed the design of the system as described in the SAR, decreased the margin of safety, and had a potentially adverse effect on the design of the system. Specifically, the screening questions addressed by the NRC ask "Does the activity change the facility as described in the SAR?" and "Could the activity adversely affect any system, structure, or component that is necessary to operate the plant in accordance with the SAR?".

The auxiliary feedwater system design function as described in the SAR is to assure sufficient feedwater supply to the steam generators, in the event of loss of the Condensate/Feedwater System, to remove energy stored in the core and primary coolant. The system may also be required in some other circumstances such as evacuation of the main control room or cooldown after a loss-of-coolant accident for a small break, including maintaining a water level in the steam generators following such a break. The SAR also contains the requirement that the structural integrity of the system be maintained. Thus, in order to answer the screening question it must be determined if the structural integrity of the system is affected or the margin of safety reduced.

NSD 209 defines the margin of safety as the margin between the design limit and the acceptance limit, with regard to the integrity of fission product barriers. The design limit is equated to the failure point, the point at which the item in question is incapable of performing it's design function. The acceptance limit is the value at which the confidence level in the integrity of the item decreases. The Catawba SAR includes IE Bulletin 79-02 which describes an anchor bolt safety factor of four as a final design and installation objective, but allows a factor of two for interim operation provided that a justification of the lower value shows that the system will remain operable until the full safety factor is restored. Catawba's accepted response to IE Bulletin 79-02 also provides that supports designed and installed prior to the effective date of the bulletin are acceptable with a safety factor of two. From this it is concluded that the acceptance limit with regard to concrete anchors is a final safety factor of four, with a lesser interim value acceptable based on an evaluation that shows operability will be maintained until the final safety factor is restored. Catawba has shown that operability has been maintained with a safety factor of two until the anchor bolts can be modified (or shown by final analysis to maintain a final safety factor of four) at the next refuel outage, therefore the margin of safety has not been reduced by the change in setpoint temperature.

Changing the alarm setpoint to 250° F and operating with a safety factor of two on the concrete anchor bolts does not reduce the margin of safety for the supports or the system. The structural integrity of the system is maintained and the margin of safety with regard to the integrity of fission product barriers is not affected. The procedure change does not reduce the flows to the steam generators, or change the timing of the flows. The auxiliary feedwater system remains capable of fulfilling its design function for all normal and accident conditions. Neither the function nor the design of the system has been changed by the activity in question, therefore negative responses to the screening questions are appropriate.

Response to Finding 2

Finding 2 states that Procedure OP/1/A/6250/02 was changed based on uncertified calculations. This finding is based upon the NRC's interpretation of EDM-101, which was assumed to be the governing directive for the evaluation in question.

The true governing directive for the evaluation is NSD 209, which addresses 10 CFR 50.59 evaluations. This directive stipulates the level of review and documentation required to perform the screening process for USQ applicability. If the activity is screened, the only documentation required is a justification of the screening. The screening form is completed and the screening questions answered by the 10 CFR 50.59 preparer, with the activity description and justification for the answers attached. The form is reviewed and signed by a Qualified Reviewer. The evaluation is then reviewed by the appropriate station supervision as part of the procedure revision process.

In the case in question the preparer of the 10 CFR 50.59 evaluation responded to the screening questions and documented the justification for the screening as required. The pipe stress analysis and support load review were utilized to provide a higher level of confidence in the conclusions drawn with regard to impact on safety margin and system function, and were not formally part of the justification. However, by signing the NSD 209 screening form and justification the preparer and Qualified Reviewer verified that all applicable inputs had been adequately performed and reviewed. It is the responsibility of these individuals to ascertain the adequacy of the inputs. There is no requirement in NSD 209 for all such data to be independently documented.

In this instance the following process was utilized. The pipe stress analysis models were analyzed using the elevated temperatures. From this analysis, support load information was generated for review. This involved the analysis of 10 separate models, and the review of over 500 supports. Due to the number of supports involved, and the nature of the support review, this effort was the most resource intensive aspect of the evaluation. Each of the models was reviewed independently for support load impact. A summary calculation was originated (in draft form) for each of the models to verify the potential affect on the supports. These calculations were then checked by an independent verifier to confirm the accuracy of the evaluation, including all assumptions and conclusions. These draft calculations were then reviewed by supervision for format, content, methodology, and technical accuracy. At this point there were a small number of non-technical editorial and format inconsistencies noted, but none that affected the conclusions of the operability evaluations. The supervisor then provided documentation to the system engineer that the operability evaluation was complete to the point of providing assurance of system operability for a setpoint of 250° F. The system engineer reviewed the analysis data, then prepared the 10 CFR 50.59 evaluation and initiated the change to the operating procedure. This change was approved February 21, 1996.

The formal calculations for the support review were re-worked to incorporate the editorial and format changes noted in the verification and supervisory review process. Since the calculations were in word processor and electronic spreadsheet format, the corrected calculations were reprinted in their entirety and verified by the original checker against the original checked copies. After the calculations were completed and signed, the original checked copies were discarded. Due to complications and emerging priorities stemming from a Loss Of Offsite Power (LOOP) event that occurred in this time frame, it required up to eight working days to complete the changes and obtain all of the appropriate signatures involved. It is for this reason that the calculations cited by the NRC were not signed and dated until March 5, 1996. This process does not violate guidance provided in NSD 209.

The EDM-101 directive is intended for calculations utilized to qualify the design and fabrication of items for construction and erection, rather than evaluation of ongoing plant operations. The EDM-101 format was used as a convenient format to finalize the documentation of the engineering review performed, but was not the governing directive. The review by the 10 CFR 50.59 preparer and Qualified Reviewer and their signing of the required NSD 209 form constituted the required documentation.

From the above discussion it is evident that the appropriate levels of review were performed prior to changing the procedure to address the abnormal conditions due to check valve backleakage. The entire process was controlled and overseen by management through the appropriate station directives, including monitoring through the PORC process described in NSD 308, and the Top Equipment Problem Review (TEPR) process as directed by Site Directive 3.0.19.

2. **Corrective Actions Taken and Results Achieved**

Station modifications have been initiated and approved which will increase the long term design temperature of the auxiliary feedwater system, and restore a safety factor of four for all concrete anchor bolts. These modifications will provide for a more detailed rigorous review of each support qualification, and physically modify those supports that do not meet the long term criteria of a safety factor of four for concrete anchors. These modifications will be implemented prior to completion of the next refueling outage for each unit. Unit 1 modifications will be completed prior to startup from 1EOC9, which is currently in progress. Unit 2 work will be performed during 2EOC8 refueling outage, currently scheduled for the first quarter of 1997. These modifications include the addition of pressure gauges to allow for better monitoring of the system operating data and valve leakage. These data will then be used in conjunction with testing to determine what further valve modifications, if any, are required. These items are being tracked by management and NRC through the station TEPR process.

Problem Investigation Process (PIP) 0-C96-1286 was generated as the tracking document for this item.

3. **Corrective Actions to be Taken to Avoid Future Violations**

No corrective actions beyond those listed in 2. above will be taken.

4. **Date of Full Compliance**

Duke Power Company is in full compliance.