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DUKE POWER

November 8, 1994

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

Subject: Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 50-414
Reply to Notice of Violation
Inspection Report Nos. 50-413/94-17 and 50-414/94-17

Gentlemen:

Attached are Duke Power Company's responses to the five (5) Level IV violations cited in the Notice of Violation of Inspection Report 50-413/94-17 and 50-414/94-17, dated September 9, 1994, for which responses are required. For those cases where a violation involved multiple examples, the associated response addresses the violation first from an overall perspective, then each example is addressed individually. Please note that Catawba is denying those violations/examples designated as A1 (only the portion of A1 pertaining to instrument inaccuracy is being denied), A2, A3, B2, B4, C, E, and F2. The basis for Catawba's denial of these violations/examples is presented in the associated response.

In addition, please find attached a discussion concerning Catawba's plans relative to service water system chemical treatment. This is in response to the NRC request for Catawba to address this issue, as contained in the cover letter to the subject inspection report.

Should you have any questions pertaining to this response or should you wish to discuss this matter further, please call M.E. Patrick at (803) 831-3681 or J.S. Forbes at (803) 831-3203.

Very truly yours,

D.L. Rehn

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Attachment

xc: S.D. Ebneter, Regional Administrator
Region II

R.J. Freudenberger, Senior Resident Inspector

R.E. Martin, Senior Project Manager
ONRR

bxc:	ELL	EC05O
	Z.L. Taylor	CN01RC
	J.E. Snyder	MG01RC
	J.E. Burchfield	ON03RC
	B.J. Horsley	EC12T
	NSRB Staff	EC12A
	NCMPA-1	
	NCEMC	
	PMPA	
	SREC	
	W.R. McCollum	CN01SM
	T.P. Harrall	CN03MA
	W.H. Miller	CN02OP
	J.S. Forbes	CN01EG
	T.E. Crawford	CN03SE
	S.W. Brown	CN03SE
	T.B. Bright	CN03MC
	D.R. Kulla	CN03MC
	A.S. Bhatnagar	CN03ES
	R.E. Hardin	CN03ES
	J.W. Cox	CN03ES
	W.J. McCabe	MG03C1
	R.P. Colaianni	EC12R
	Master File CN-815.01	
	IR File 94-17	

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-02

Notice of Violation

A. 10 CFR 50, Appendix B, Criterion III, "Design Control" requires that "Measures shall be established to assure that applicable regulatory requirements and the design basis ... are correctly translated into specifications, drawings, procedures, and instructions."

Contrary to the above, as of August 1, 1994, applicable design basis had not been correctly translated into specifications, drawings or procedures in that:

1. The ultimate heat sink analysis did not consider pump heat, inventory loss via seepage or the fire protection, auxiliary feedwater, component cooling water and fuel pool makeup systems, level and instrument inaccuracies causing the theoretical peak temperature of 100°F to be exceeded by 0.5°F.

2. Calculation CNC-1223.24-00-0001, "Catawba Nuclear Station - Unit 1 & 2 Size Nuclear Service Water Discharge Short Leg To Standby Nuclear Service Water Pond Flow Restrictor," did not use a piping resistance factor consistent with the pipe's service environment.

3. Calculation CNC-1223.24-00-0013, "Nuclear Service Water System Design Verification," did not validate select heat load assumptions, use the maximum allowable inlet temperature for the component cooling water heat exchangers, size the emergency diesel generator starting air aftercooler and component cooling water heat exchanger relief valves such that their relieving capacity would keep system pressure less than or equal to system design pressure, or use Final Safety Analysis Report auxiliary feedwater flows of 900 gallons per minute.

4. Revisions to design document CNTC-1574-RN-S002 did not designate a change to the low flow setpoint of alarm response procedure OP/1&2/A/6100/10M or emergency procedure, EP/1&2/A/50/ES-1.3, "Aligning NS for Recirculation," such that normal containment spray heat exchanger flow was less than those indicated in the alarm response procedure or the emergency procedure.

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

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-02

RESPONSE: (General)

1. **Reason for Violation**

10 CFR 50, Appendix B, Criterion III, "Design Control," requires that the design bases and regulatory requirements for the structures, systems, and components of the facility be translated into specifications, drawings , procedures, and instructions. These documents are to be controlled in a manner that ensures that they are correct and that allows no deviations from the standards.

Three examples of analysis deficiency and one example of a document deviation were identified. Two of the analysis deficiency examples are being denied in their entirety and the third is being partially denied, as described in the subsequent responses to the specific violation examples. These violation examples were all related to design documents and the process of reviewing design documents both within the engineering and plant organizations.

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

Reviews have been conducted to ensure that the violation examples noted do not compromise the ability of the affected structures, systems, or components to fulfill their safety-related functions or in any way reduce the level of safety of the station.

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

Refer to the specific corrective actions described in the responses to Violation Examples A1 and A4.

4. **Date of Full Compliance**

Document review and revision will be complete as stated in the specific responses to Violation Examples A1 and A4.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-02

RESPONSE: (Example A1)

1. **Reason for Violation**

The ultimate heat sink analysis did not consider pump heat, inventory loss via seepage or the fire protection, auxiliary feedwater, component cooling water and fuel pool makeup systems, level and instrument inaccuracies, causing the theoretical peak temperature of 100°F to be exceeded by 0.5°F. Note that Catawba is denying the portion of this example pertaining to the consideration of instrument inaccuracy. The basis for this denial is fully explained in the response to Violation E.

This example has been attributed to analysis deficiency, especially with regard to documentation of assumptions and how those assumptions impact the rigor and conservatism of the analysis.

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

To account for the additional heat load and also the inaccuracy of instruments used to monitor SNSWP parameters (Catawba is denying that portion of this violation example pertaining to the consideration of instrument inaccuracy; refer to Catawba's response to Violation E, which Catawba is also denying, for a complete discussion of instrument inaccuracy), the SNSWP temperature and level technical specification surveillance has been changed from 91.5°F and 570 feet to 88°F and 570.2 feet. These changes ensure that the service water supply temperature assumed in the containment pressure analysis (92°F) and the design temperature assumed for long-term RN pump motor and diesel generator operation (100°F) will not be exceeded.

Documentation is being completed in the form of a Past Operability Determination that addresses the additional heat load, inventory loss, instrument inaccuracy, and the effect on the SNSWP analysis and station operation prior to the implementation of the new surveillance limits mentioned above. The additional heat load from pump work, inventory losses from seepage and safety system makeup, and instrument inaccuracies have been determined. While the documentation has not been completed, the analysis has been performed with the revised heat load, inventory, and initial starting conditions. These changes do not significantly impact the SNSWP analysis. The worst case short term supply temperature, which affects peak containment pressure and temperature, was determined to be 94.5°F. A review of the heat transfer calculations used to support the Catawba peak containment pressure analysis has shown that sufficient heat removal capability was present at this higher service water temperature. It has been verified that the long term supply temperature, assumed to be 100°F, would have remained below 97.5°F.

Meteorological data from 1993, rather than the design basis meteorology, was used in this analysis to reflect the worst case conditions which have been encountered since startup.

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

The design calculations associated with the SNSWP analysis are being revised. CNC-1223.24-00-0006, "Nuclear Service Water System HX Outlet Temperature Calculation and Heat Load Rejected to SNSWP," is being revised to more carefully examine all heat loads, including pump work, core decay, and the sensible (cooldown) heat of containment and systems. CNC-1223.24-

00-0013, "Nuclear Service Water System Design Verification," is being revised to provide as inputs to the SNSWP analysis any inventory losses related to service water makeup to safety related systems. CNC-1150.01-00-0001, "Standby Nuclear Service Water Pond - Thermal Analysis During One Unit LOCA and One Unit Shutdown," will be revised based on the inputs from the heat load and inventory calculations mentioned above. This analysis will determine whether or not CNS can return to the surveillance practice based on technical specification limits of 91.5°F initial temperature and 570 feet surface elevation.

4. **Date of Full Compliance**

The above referenced calculations will be revised by April 1, 1995.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-02

RESPONSE: (Example A2)

1. **Basis for Denying Violation**

It was the inspection teams opinion that Calculation CNC-1223.24-00-0001, "Size Nuclear Service Water Discharge Short Leg to Standby Nuclear Service Water Pond Flow Restrictor," did not use a piping resistance factor consistent with the pipe's service environment at the time the calculation was originated.

This example of the proposed violation takes exception to an engineering decision made during the original plant design effort. Catawba is denying this violation example. It is Catawba's position that the above calculation was not deficient.

The above calculation was written in 1976, prior to the construction of the nuclear service water system. The purpose of the calculation was to size a flow restricting orifice to evenly distribute nuclear service water discharge flow to both discharge legs of the standby nuclear service water pond (SNSWP).

The calculation references the 1970 version of Ingersoll-Rand Pump Company's "Cameron Hydraulic Data" as the source for the flow and pressure drop correlations used in determining friction losses in pipe. This reference uses the widely-accepted Hazen and William's empirical formula for friction loss. The formula includes a constant accounting for surface roughness. For steel pipe, the range of values given for the roughness constant is 80 to 150, where the low value corresponds to poor pipe and 150 corresponds to smooth pipe.

In the calculation to size the flow restricting orifice, a roughness constant of 100 was chosen. The inspection team believed that the lower constant of 80 should have been used in 1976.

Catawba maintains that the above described calculation was not deficient. The pressure drop correlations used and the piping condition assumed constituted the best information available at the time. Calculational input decisions were made as a result of Duke Power Company's accumulated experience with service water system performance at both nuclear (Oconee) and fossil stations. Should the calculation be performed today, the roughness constant of 100 would still be used for large diameter piping, based upon existing knowledge of the pipe's service environment.

The calculation that used this roughness constant is a stand-alone calculation used specifically to size the short leg flow restricting orifice. This constant was not applied in any other calculations for the RN system other than for preliminary sizing which was subsequently verified through pre-operational and periodic testing.

As a check, a recalculation of the orifice size using current methodology and validated assumptions confirmed the results of the original calculation. The orifice size developed in the recalculation is essentially the same as was developed in the original calculation. The design basis (i.e., even flow split) is unaffected.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-02

RESPONSE: (Example A3)

1. **Basis for Denying Violation**

Calculation CNC-1223.24-00-0013, "Nuclear Service Water System Design Verification," was originated in 1984 to verify:

1. The design parameters (as shown on system flow diagrams) of QA Condition 1 pipe, including Duke pipe class, pipe material, design temperatures, and design pressures of the nuclear service water system,
2. The ability of the system to deliver the required flow to QA Condition 1 systems and equipment, and
3. The QA Condition 1 instrumentation setpoints and interlocks.

The calculation shows that the design conditions assigned to the system are such that they are not expected to be exceeded during any design basis mode of operation, with high levels of conservatism built in due to very conservative assumptions made in the analysis. In most cases, the approach taken in the calculation was to superimpose all design conditions on the system to show that the assigned design pressure and temperature values would not be exceeded. This approach was conservative in that the simultaneous combinations of loads on the system were often beyond the system design basis load combinations. This was done during the initial design phase. Subsequently, as revisions are made, instances are encountered where actual design basis load combinations are used. Such was the case in this example.

Calculation did not validate select heat load assumptions:

10 CFR 50, Appendix B, Criterion III, "Design Control," requires that design control measures provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. The referenced calculation meets this requirement in that it was verified by a qualified reviewer and approved, thereby satisfying the requirements of 10 CFR 50, Appendix B, Criterion III.

The exact heat load of several small components cooled by nuclear service water was not known when the calculation was originated. Based on the experience of power plant engineering, these loads were assumed to be small enough that the nuclear service water exit temperature would be less than the design temperature of the piping, 150°F. The engineer who performed the calculation was familiar with the nuclear service water system and with the components being reviewed. These assumptions were validated by the reviewer, thereby meeting the requirements of 10 CFR 50, Appendix B, Criterion III. Operating experience has shown that the nuclear service water exit temperature is well below 150°F in each case and the assumptions made in the calculation have been determined to be appropriate.

The components for which the assumptions of a < 50°F ΔT were made are:

Instrument air compressors and aftercoolers
Reciprocating charging pump fluid drive cooler
Reactor coolant pump motor coolers

These components are not safety-related and are supplied by the nuclear service water non-essential headers. The non-essential headers are isolated by either a high-high containment pressure or nuclear service water suction realignment to the standby nuclear service water pond (SNSWP). The nuclear service water non-essential headers, components, and heat loads are not significant to the system design basis. The assumptions made in the calculation were conservative and appropriate.

Calculation did not use the maximum allowable inlet temperature for the component cooling water heat exchangers:

The calculation did in fact use the maximum allowable inlet temperature in the design basis analysis. The component cooling (KC) heat exchanger overtemperature condition noted in this beyond design basis heat load calculation superimposed the heat load of a non-LOCA unit fast cooldown with the minimum nuclear service water flow from a one-pump event to arrive at the highest conceivable nuclear service water exit temperature. This combination is beyond the system design basis since a one-pump event is only valid with the non-LOCA unit in Mode 5 (cold shutdown). The section of the calculation containing the design basis analysis was located **after** the section containing the beyond design basis analysis and **did** use the maximum allowable inlet temperature. This calculation did produce acceptable results.

Calculation did not size the emergency diesel generator starting air aftercooler and component cooling water heat exchanger relief valves such that their relieving capacity would keep system pressure less than or equal to system design pressure:

The nuclear service water system was designed to the 1974 edition of the ASME Boiler and Pressure Vessel Code. Section III, Division I, Article ND-7000, "Protection Against Overpressure," states:

> "Vessels, tanks, piping, pumps, and valves shall be protected while in service from the consequences arising from the application of steady state or transient conditions of pressure and coincident temperature that are in excess of the design conditions specified for the system."

The diesel generator starting air aftercoolers are considered to be unisolable while in service. The calculation explains that the only way for an in-service diesel generator starting air aftercooler to be isolated from the nuclear service water discharge would be a multiple failure of various locked-open or electrically interlocked isolation valves. Therefore the requirements of the code are met. If isolated while removed from service, the only overpressure event that would exceed the capacity of the relief valves would be a tube rupture in the aftercooler. Even if this were to occur, the aftercooler is separated from the starting air receiver tank by two in-series check valves. Therefore, the overpressure protection in this situation would only have to relieve an extremely short run of pressurized piping. The overpressure relief protection installed at the diesel generator starting air aftercoolers is considered to be adequately sized to meet design basis and Code requirements.

These multiple failures are beyond the design basis of the system. The calculation was reviewed and approved by a qualified individual and satisfies 10 CFR 50, Appendix B, Criterion III.

Likewise, the KC heat exchangers are also not considered to be isolable while in service. It is Catawba's position that the relief valves provided are sufficient to meet Code requirements. The calculation states that the nuclear service water piping which supplies the KC heat exchangers could be isolated by a combination of both a failure that would result in the closure of the discharge isolation valve and operator error resulting in failure to open the inlet valve when placing the KC heat exchanger in service. The calculation recommends that steps be taken to ensure that the nuclear service water inlet valve be open whenever a KC heat exchanger is in service. Procedural guidance is in place to ensure that the nuclear service water inlet isolation valve is open whenever a KC heat exchanger is in service. A KC heat exchanger is inoperable if the nuclear service water inlet isolation valve to that component is not open. This is administratively controlled by Operations procedures OP/0/A(B)/6400/06C, "Nuclear Service Water System," PT/1/A(B)/4400/02C, "Nuclear Service Water Valve Verification," and AP/1(2)/A/5500/019, "Loss of Residual Heat Removal System." In addition, each KC train window on the control room 1.47 bypass panel will illuminate "BYPASSED" anytime the nuclear service water inlet isolation valve to the corresponding KC heat exchanger is not open. These measures are adequate to prevent isolation of the KC heat exchangers.

Calculation did not use Final Safety Analysis Report auxiliary feedwater flows of 900 gallons per minute:

The calculation intentionally used conservative values, rather than nominal values specified in the FSAR. The calculation determined the nuclear service water pressure available to supply assured makeup to the auxiliary feedwater (CA) system. Flows to the Unit 1 and 2 CA systems were assumed to be 1500 and 1900 gpm, rather than the 900 gpm stated in FSAR Table 9-3. The calculation demonstrates that if the system were capable of delivering 1500 and 1900 gpm for the given pressure at the RN to CA interface, then the 900 gpm specified in the FSAR would certainly be delivered.

Because the values used in the analysis were unquestionably conservative, the calculation is sufficient for determining the nuclear service water pressure available to supply the CA system.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-02

<u>RESPONSE</u>: (Example A4)

1. **<u>Reason for Violation</u>**

Revisions to design document CNTC-1574-RN-S002 did not designate a change to the low flow setpoint of alarm response procedure OP/1(2)/A/6100/10M or emergency procedure EP/1(2)/A/50/ES-1.3, "Aligning NS for Recirculation," such that normal containment spray heat exchanger flow was less than those indicated in the alarm response procedure or the emergency procedure.

This example has been attributed to changes not adequately communicated. The process that was in place for changing test acceptance criteria did not include a review by station groups for impact to their procedures.

2. **<u>Corrective Actions Taken and Results Achieved</u>**

PIP 0-C94-1387 has been initiated to correct the discrepancy between the design documents and plant operating parameters.

3. **<u>Corrective Actions to be Taken to Avoid Future Violations</u>**

A change process has been developed that will require all station groups to review any change to design documents. This "editorial change" process is waiting on approval for implementation.

4. **<u>Date of Full Compliance</u>**

Catawba Nuclear Station will be in full compliance by December 28, 1994.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-05

Notice of Violation

B. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Catawba maintenance procedure, MP/0/A/7150/98, "Nuclear Service Water (RN) Pump Bearing Injection, Stuffing Box, Oil and Motor Cooler Lines Flushing and Chemical Cleaning," provides the direction on how to flush service water motor coolers.

Catawba periodic test, PT/1/A/4400/06E, "KD Heat Exchanger 1A Heat Capacity Test," states in step 9.0 under test method that "The heat exchanger under test will have shell side and tube side flow set up as close to design flow as possible. Inlet and outlet temperatures will be taken for both sides of the heat exchanger. From this data a tube side fouling factor will be obtained."

Section 4.1.2, "Slope," of drawing ICS-A-20.2, Revision 8, September 8, 1992, "Instrument Standards, Installation Field Practices," section 4.2.3, "Expansion Loops," of drawing ICS-A-20.04-01, Revision 15, June 8, 1989, "Instrument Standards, Installation Field Practices," and notes on instrument detail drawings CN-1499-RN56, Revision 4, April 13, 1984, and CN-2499-RN56, Revision 3, September 27, 1983, both titled "NSW Pump Strainer D/P," and CN-2499-RN3, Revision 8, September 27, 1983, "RN Pump Motor Cooler Outlet Flow," require a continuous downward slope from the instrument tap to the instrument, a continuous downward slope from the vent to the instrument line and S-type expansion loops for service water strainer and motor cooling instruments.

Catawba 10 CFR 50.59 Screening Checklists require the completion of a safety evaluation when the answer to any of the questions asked is yes. One of the questions states, "Does this evaluation item affect structures, systems or components that are addressed in the Final Safety Analysis Report in a significant manner?"

Contrary to the above:

1. On July 4, 1994, an activity affecting quality was not accomplished in accordance with prescribed procedures in that a chemical flush of the 1A RN pump was not accomplished using MP/0/A/7150/98, "Nuclear Service Water (RN) Pump Bearing Injection, Stuffing Box, Oil and Motor Cooler Lines Flushing and Chemical Cleaning," Revision 0, February 4, 1992, in that maintenance personnel flushed only a portion of the piping using vendor supplied hose connections rather than flushing the coolers as specified by the procedure.

2. An activity affecting quality was not accomplished in accordance with prescribed procedures in that a flow in excess of 1400 gallons per minute instead of the design flow of 900 gallons per minute was achieved by marking as not applicable steps 12.4, 12.5 and 12.6 which throttle the tube side flow in completed PT/1/A/4400/06E procedures dated October 14, 1992 and August 17, 1993.

3. As of August 1, 1994, an activity affecting quality was not accomplished in accordance with prescribed drawings in that the high pressure side instrument lines for nuclear service water pump strainers 1A, 1B, 2A, 2B were not S-type or continuously sloped down from the instrument tap to the instrument and the instrument line for nuclear service water pump 2B motor cooler flow element, 2RNFE-7410, was not continuously sloped downward from the vent to the instrument.

4. In 1990, an activity affecting quality was not accomplished in accordance with prescribed instructions in that the question, "Does this evaluation item affect structures, systems or components that are addressed in the Final Safety Analysis Report in a significant manner?," to the 10 CFR 50.59 checklist for Exempt Change CE-3137 was answered no instead of yes even though the component being modified, cooling water outlet piping from the upper bearing oil cooler for safety related service water pump motor 1A, was described in the Final Safety Analysis Report.

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

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-05

RESPONSE: (General)

1. **Reason for Violation**

The four examples cited in this violation as failure to perform activities affecting quality in accordance with prescribed procedures encompass diverse station activities. Catawba is denying violation examples B2 and B4.

The first example deals with failure to follow technical procedures and is therefore strictly a procedure adherence issue. It is station management's expectation that procedures be strictly adhered to and this incident clearly did not meet management expectations.

The second example is being denied. The procedure section addressed in the violation example is an abstract (in paragraph format) of the test method that will be followed. This portion of the procedure should have been updated to accurately reflect the procedure steps that are currently in place to test the diesel generator engine cooling water heat exchanger at off-design flow conditions; however, Catawba maintains that this does not constitute a violation.

The third example identified the need to correct a design drawing to reflect the as-built configuration of instrument tubing that was determined to be installed correctly. This also identified the need to reemphasize to station personnel that instrument tubing is fragile and is not to be used as foot holds or to facilitate climbing.

The fourth example is being denied. Catawba maintains that the program in place for conducting and reviewing 10 CFR 50.59 evaluations is adequate.

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

Specific corrective actions are addressed for each example not being denied.

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

Specific corrective actions are addressed for each example not being denied.

4. **Date of Full Compliance**

Catawba Nuclear Station is in full compliance.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-05

RESPONSE: (Example B1)

1. **Reason for Violation**

The reason for this violation example (motor cooler chemical flush) is failure to follow procedure.

On July 4, 1994, it became necessary to perform a flush of the motor cooler of RN pump 1A. Maintenance procedure MP/0/A/7150/98 provides the guidance to conduct the flush. In reviewing the flush method described in the procedure, the engineer concluded that the method would not be as effective as desired. Therefore, the engineer specified an alternate flush connection on the work order in lieu of revising the procedure. This is clearly not in accordance with management expectations on procedure use and adherence.

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

A revision to Maintenance procedure MP/0/A/7150/98 was issued on August 3, 1994 to describe the preferred method of flushing, which was the method used on July 4, 1994.

The Engineering and Maintenance personnel involved in this work activity have been counseled.

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

The corrective steps that have been taken as noted above are sufficient to prevent future violations. Management is continuing to aggressively follow up on all procedure use and adherence issues.

4. **Date of Full Compliance**

Catawba Nuclear Station is in full compliance.

RESPONSE: (Example B2)

1. **Basis for Denying Violation**

PT/1/A/4400/06E provides procedural guidance for marking steps 12.4, 12.5, and 12.6 "not applicable". These three steps are IF, THEN steps and therefore are only required to be performed under certain conditions as explained below. In this case, the conditions were not met and the steps were allowed to be marked "not applicable". Marking these steps as "not applicable" was in conformance with the procedural requirements and did not violate the Duke Power Company procedure adherence policy.

The following procedures control the performance of the emergency diesel generator engine jacket water cooler (KD) heat capacity tests:

PT/1/A/4400/06E, "KD Heat Exchanger 1A Heat Capacity Test"
PT/1/A/4400/06F, "KD Heat Exchanger 1B Heat Capacity Test"
PT/2/A/4400/06E, "KD Heat Exchanger 2A Heat Capacity Test"
PT/2/A/4400/06F, "KD Heat Exchanger 2B Heat Capacity Test"

Section 9 of these tests, titled "Test Method", is a descriptive section. Section 9 does not contain any instructions, steps, or acceptance criteria. The statement referenced in the violation is only found in Section 9 of the tests. The user is expected to strictly adhere to Section 12 of the procedures.

Section 12 of these tests, titled "Procedure", contains the steps to be completed in performing the heat capacity test. Section 12 was correctly adhered to as expected in the execution of this procedure. While Catawba does not consider this incident to be a procedure adherence issue, it is nevertheless recognized that the procedure was deficient from a human factors standpoint in that Section 9 was not in complete agreement with Section 12. The "Test Method" sections of these procedures have been revised to accurately reflect the procedure steps.

The temperature of the diesel generator cooling water is controlled by a three-way modulating valve. This valve maintains the temperature exiting the diesel generator at 165°F by allowing some portion of the coolant to bypass the KD heat exchanger. During winter months, when RN temperatures are around 50°F, the valve controller has a tendency to overcorrect, or "hunt", for the flow split required to maintain 165°F.

Performance of the heat capacity tests requires stable flows and temperatures on both sides of the heat exchanger. The steps in Section 12 of the KD heat exchanger tests allow the RN flow to be manually throttled if the modulating valve is not able to stabilize on its own. A note prior to these steps allows them to be marked as "not applicable" if manual throttling is not required for flow stability. Unnecessary throttling of the RN flow to the KD heat exchangers is undesirable. The throttle valve position is set and locked during the RN flow balance. When this position is changed, the diesel generator and any system supported by the diesel generator are considered to be inoperable.

Flow coefficient equations are used to calculate the fouling factor and to correct back to design conditions. Flow rates above or below the design flow do not invalidate the equations as long as

the Reynolds number is within the acceptable range. For each data point, the Reynolds number is calculated and verified to be in the acceptable range.

It is station management's expectation that procedures be strictly adhered to and if they cannot, then work is to be stopped and supervision contacted to correct the discrepancy. It is management's position that these procedures are being correctly followed as stated above.

DUKE POWER COMPANY
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50-413, 414/94-17-05

RESPONSE: (Example B3)

1. **Reason for Violation**

The reason for the instrument lines on the nuclear service water pump strainers 1A, 1B, 2A, 2B and B pump motor cooler outlet flow not meeting slope requirements is due to damage caused by personnel stepping on the lines and/or using the lines as hand holds.

In order to determine the reason for the expansion coils for nuclear service water pump strainers 1A, 1B, 2A and 2B not being S-type design coils, a search was performed to determine if any work was conducted that would have replaced these loops with the circular design. No work was found that affected these loops. The acceptability of the substitution of the circular coils for the S-type coils can be determined based on the information included on the drawing. Therefore, it is believed that these loops were originally installed as circular loops based on the installer's interpretation of the information provided on the drawing. Furthermore, it was an oversight that the drawing was not revised to clearly reflect the acceptability of this substitution.

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

The following work orders were completed to correct the slope for the expansion coils and all slope requirements were met.

- W/O 9405923801 - 2RNPG7501
- W/O 9405924901 - 1RNPG7501
- W/O 9405923301 - 2RNPG7491
- W/O 9405924401 - 1RNPG7491
- W/O 9405924701 - 2RNPG7410

PIP 0-C94-1136 was generated to address why the type of expansion coils found in the field appeared to be different from what is referenced in the installation drawing, CN-1499-MI44.00, for these loops. Per this PIP, the actual installation exceeds the requirements of the design specified because it provides more movement than the S-type design listed on the drawing. Drawing CN-1499-MI44.00 was clarified by deleting loops RN7490 (including 7491) and 7500 (including 7501) from the exception list which specified the S-type design. This correction to the drawing was made per minor modification CE-4679.

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

It will be continually emphasized to station personnel concerning items that are not meant to be stepped on (i.e., snubbers, small pipes, instrument lines, etc.). Also, existing Site Directive 3.11.3 stresses that those items which are not capable of supporting personnel should never be used for climbing. It has been stressed to personnel to take appropriate measures as needed to avoid damage to tubing, snubbers, etc., and to be proactive if inadvertent damage does occur to ensure that the component is repaired. In addition, the annual General Employee Training includes discussions on each individual's responsibility for the protection of station equipment. Also, signs are posted throughout the plant, stressing what can and cannot be used as a climbing tool. All of these steps have been and are still in existence. Finally, IP/0/A/3890/03, "Instrument

Tubing Installation Procedure," has been revised to include a note stressing slope requirements and referencing the necessary drawings and standards to meet these requirements.

4. **Date of Full Compliance**

Catawba Nuclear Station is in full compliance.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-05

RESPONSE: (Example B4)

1. **Basis for Denying Violation**

It is Catawba's position that under the 10 CFR 50.59 process in place at the time of the referenced evaluation, the USQ screening question, "Does this evaluation item: affect structures, systems, or components that are addressed in the FSAR in a significant manner?", was appropriately checked "No". Catawba, therefore, considers the 10 CFR 50.59 checklist screening questions for Exempt Change CE-3137 to have been answered correctly and is denying this violation example.

Exempt Change CE-3137 was generated to revise the piping ISO drawing, CN-1492-RN234, to allow 3/8" schedule 40 pipe to be substituted for 3/8" schedule 80 pipe. The section of piping affected was the discharge connection to the upper bearing oil cooler for RN pump motor 1A.

The screening for 10 CFR 50.59 applicability check list Part 3 was checked "No", including the item, "Does this evaluation item: affect structures, systems, or components that are addressed in the FSAR in a significant manner?".

Exempt Change CE-3137 only revised a segment of piping; a segment of piping is considered a component of the system. This component of the RN system is not addressed in the FSAR, nor was this change considered significant. The change did not affect either the system function or any other component. The subject change involved the RN system and this system is addressed in the FSAR. Section 9.2.1.2.3 of the FSAR addresses three items relative to the RN pump motor upper bearing oil coolers:

- The conditions under which they are supplied cooling flow
- The relative location with respect to the associated pump's backflush RN strainer
- The valve alignment to the pump start/stop conditions

Exempt Change CE-3137 was generated to update the piping ISO drawing. Like-for-like pipe was changed that met the design basis. The change to this component (the segment of piping) was not significant. The piping specification, CNS-1206.00-02-1002, permits schedule 40 pipe to be used if design conditions are not greater than 200 psig and 200°F. Design conditions for this application are 100 psig and 150°F. A routine piping analysis calculation, CNC-1206.02-84-2021, Rev. 6, was performed at the time of this modification to reflect the change in pipe schedule. This was not considered significant with respect to seismic design of this system but was necessary to maintain documents and calculations current. The exempt change was written to update the drawing to indicate the correct pipe schedule.

The component which was the subject of this exempt change (the segment of piping) was not addressed in the FSAR. Historically, the question, "Does this evaluation item: affect structures, systems, or components that are addressed in the FSAR in a significant manner?", was always answered "No" for like component changeouts that met the original design basis. The structure of Catawba's 10 CFR 50.59 program in place at the time of this modification, as well as Catawba's current program, is based on NSAC-125, "Guidelines for 10CFR50.59 Safety Evaluations."

Nuclear System Directive (NSD) 209, "10 CFR 50.59 Evaluations," was revised on June 9, 1994. This revision combined the former NSD 209, "50.59 Evaluation of Nuclear Facility Modifications," and NSD 210, "50.59 Evaluation of Nuclear Facility Procedures," into one directive. Part of this revision included the implementation of new Unreviewed Safety Question (USQ) applicability screening criteria. As a result of this NSD revision, the above described screening question no longer exists. The new appropriate USQ screening questions to answer are (1) "Does the activity change the facility as described in the SAR?" and (2) "Could the activity adversely affect any system, structure, or component necessary to operate the plant in accordance with the SAR?". It is Catawba's position that had Exempt Change CE-3137 been performed under the revised NSD 209, the above two questions would have been answered "No". Under the 10 CFR 50.59 evaluation process in place at the time, it is also maintained that Catawba was correct in answering the appropriate USQ screening question "No". Catawba, therefore, considers the 10 CFR 50.59 checklist screening questions for Exempt Change CE-3137 to have been answered correctly and is denying this violation example.

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CATAWBA NUCLEAR STATION
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50-413, 414/94-17-06

Notice of Violation

C. Technical Specification 6.8.2.8.a states, "The Nuclear Safety Review Board shall be responsible for the review of the safety evaluation for: (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR to verify that such actions did not constitute an unreviewed safety question."

Contrary to the above in 1991 the Nuclear Safety Review Board did not review a safety evaluation for replacing the lower pump bearing with a rubber cutlass-type bearing, removing the lube injection tube which surrounded the pump shaft, and removing the lower rings of packing from the packing gland and replacing them with lantern rings for the 1A safety related service water pump.

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

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-06

RESPONSE:

1. **Basis for Denying Violation**

As stated in the inspection report, the team felt that the Nuclear Safety Review Board (NSRB) did not review safety evaluations for modifications performed on the nuclear service water pumps in 1991.

The nuclear service water pumps were modified under Exempt Changes CE-3004, 3005, 3006, and 3007. The safety evaluations for these exempt changes were reviewed by the NSRB per Technical Specification 6.5.2.8a. Catawba has reviewed the safety evaluations performed for these exempt changes and considers them to be acceptable.

These modifications were designed by the pump manufacturer, Bingham, and implemented by Johnson Pump Company. The following statements are contained in the accompanying 10 CFR 50.59 evaluations for the referenced modifications:

> "RN pump (1A, 1B, 2A, 2B) will be modified such that bearing lube injection requirements are provided for by the pump itself and no external lube flows are required. The pump manufacturer, Bingham, has evaluated this modification and has concurred with this change. Design Engineering has also reviewed the change and determined that RN pump (1A, 1B, 2A, 2B) operation will not be affected."

This modification included replacing the pump bearing, removing the lube injection tube, and replacing the lower rings of packing with lantern rings for pump 1A. Duke Power Company letter MCSE-91-6 documented Design Engineering's internal review of the overall modification and its effect on the RN system.

A 10 CFR 50.59 evaluation by itself is not intended to be a stand-alone document. The exempt changes contain mark-ups from the pump manufacturer of all the affected drawings related to the changes to the pumps. The changes were performed by Johnson Pump Company under their "N" stamp program. The changes were engineered by the original equipment manufacturer (OEM), Sulzer-Bingham Pumps, and their approval is documented in the safety evaluation and by letters on file. This change did not affect the performance or operation of the RN pumps.

For situations involving modification work performed by a QA vendor under its "N" stamp program, it is Catawba's expectation that the work meet all requirements specified by Duke Power Company. It is Catawba's position that a QA approved, original equipment manufacturer possesses the technical expertise, component operating experience, and design knowledge to conduct a thorough technical evaluation of all changes to its equipment, including the adequacy of the modification (i.e., whether it resolves the original problem with the equipment) and all interactions within the equipment itself. In this case, it was expected that the bearing replacement meet all requirements for the specified pump. It is Catawba's expectation that the preparer of a 10 CFR 50.59 evaluation (a Catawba individual) review a modification for its potential impact on nuclear safety, license requirements, and design basis. When planning a modification, the originator develops specifications that are ensured to meet the system design basis, system requirements, and license requirements. These specifications are then provided to the vendor. The vendor provides documentation back to Duke Power Company that the delivered

component meets the specifications. Finally, the preparer reviews the component for overall impact on the system design basis, system requirements, and license requirements. This was done as described above. It is not Catawba's expectation for the 10 CFR 50.59 evaluation preparer to possess design knowledge particular to an engineered component, superior to that of a QA approved vendor who specializes in the design and manufacture of such components, to dispute the technical adequacy of vendor work. Duke Power Company typically does not possess proprietary vendor design history and documentation necessary to conduct such an evaluation. It is expected that the preparer address the vendor changes in a general sense in the 10 CFR 50.59 evaluation such that there is an overall representation of the vendor changes being made. Therefore, in this case, it is acceptable for the 10 CFR 50.59 evaluation to not provide explicit details concerning the pump bearing, shaft tube, and packing changes.

In this instance, all of the above expectations were met for both the vendor and the 10 CFR 50.59 evaluation preparer. On this basis, Catawba is denying the alleged violation.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
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Notice of Violation

E. Technical Specification 3.7.5 requires that the standby nuclear service water pond have a minimum water level at or above elevation 570 feet Mean Sea Level and an average water temperature of less than or equal to 91.5°F at elevation 568 feet in the pond. Technical Specification Surveillance 4.7.5.a requires a verification of the water level to be within the limit at least once per 24 hours. During the months of July, August and September, Technical Specification Surveillance 4.7.5.b requires that water temperature be verified to be within its limit at least once per 24 hours.

Contrary to the above, as of August 8, 1994, verification that acceptable minimum water level and maximum average water temperature for the standby nuclear service water pond was not assured in that the acceptance criteria for Periodic Test Procedure, PT/1/A/4600/02A and PT/2/A/4600/02A, Mode 1 Periodic Surveillance Items, were established using the limit of 570 feet for minimum level and 91.5°F for maximum temperature of the pond without accounting for instrument inaccuracies.

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

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-11

RESPONSE:

I. **Basis for Denying Violation**

Catawba denies the violation on the basis that the surveillance was appropriately performed and that instrumentation inaccuracy in this case has no significant impact on plant safety and is in accordance with published regulations.

In general, regulations do not specify how instrument inaccuracy is to be applied except for Limiting Safety System Settings (LSSS) which are governed by 10 CFR 50.36 paragraph (c)(l)(ii)(A), which states:

> "Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

A specific method for meeting the requirements of 10 CFR 50.36 paragraph (c)(l)(ii)(A) is presented in Regulatory Guide 1.105. Catawba utilizes the methodology of Regulatory Guide 1.105 for those variables for which instrument inaccuracy is to be applied. For other variables, the method of controlling instrument error for non-LSSS situations may be defined and accounted for by the licensee.

The Standby Nuclear Service Water Pond temperature and level are not Limiting Safety System Settings as defined by the Catawba Technical Specifications. These instruments fall under the general requirements of 10 CFR 50, Appendix A, Criterion 13, Instrumentation and control, which states:

> "Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables within prescribed operating ranges."

Additionally, with respect to containment functions, these instruments could also fall under the regulatory guidance of 10 CFR 50 Appendix A, Criterion 16, Containment design, which states:

> "Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

In accordance with these regulations, instrument inaccuracy at Catawba is typically applied only in those cases where it would have an impact on safety. In those cases, instrument inaccuracy is

typically accounted for in the accident analysis. This is accomplished by taking the Technical Specification value, applying an appropriate bias for instrument inaccuracy, then using that value as the starting point for the accident analysis. This approach has the obvious human factors benefit of allowing the operators to read control room or plant instrumentation and directly compare the indicated value to Technical Specifications to determine compliance without having to look through a separate surveillance procedure.

Because of this approach to controlling instrument inaccuracy, there are only a few isolated cases of it being accounted for in a station surveillance procedure. These cases are identified and controlled by Site Directive 3.2.2, "Development and Approval of the Periodic Testing Program," and typically only include surveillance values which are satisfied during system testing. Technical Specification Surveillance 4.7.5.a and 4.7.5.b are not identified in this directive and therefore, appropriately, no allowance for instrument inaccuracy is included in the surveillance procedure value.

A review has been conducted to determine the appropriateness of not including an allowance for instrument inaccuracy of the SNSWP temperature and level instrumentation.

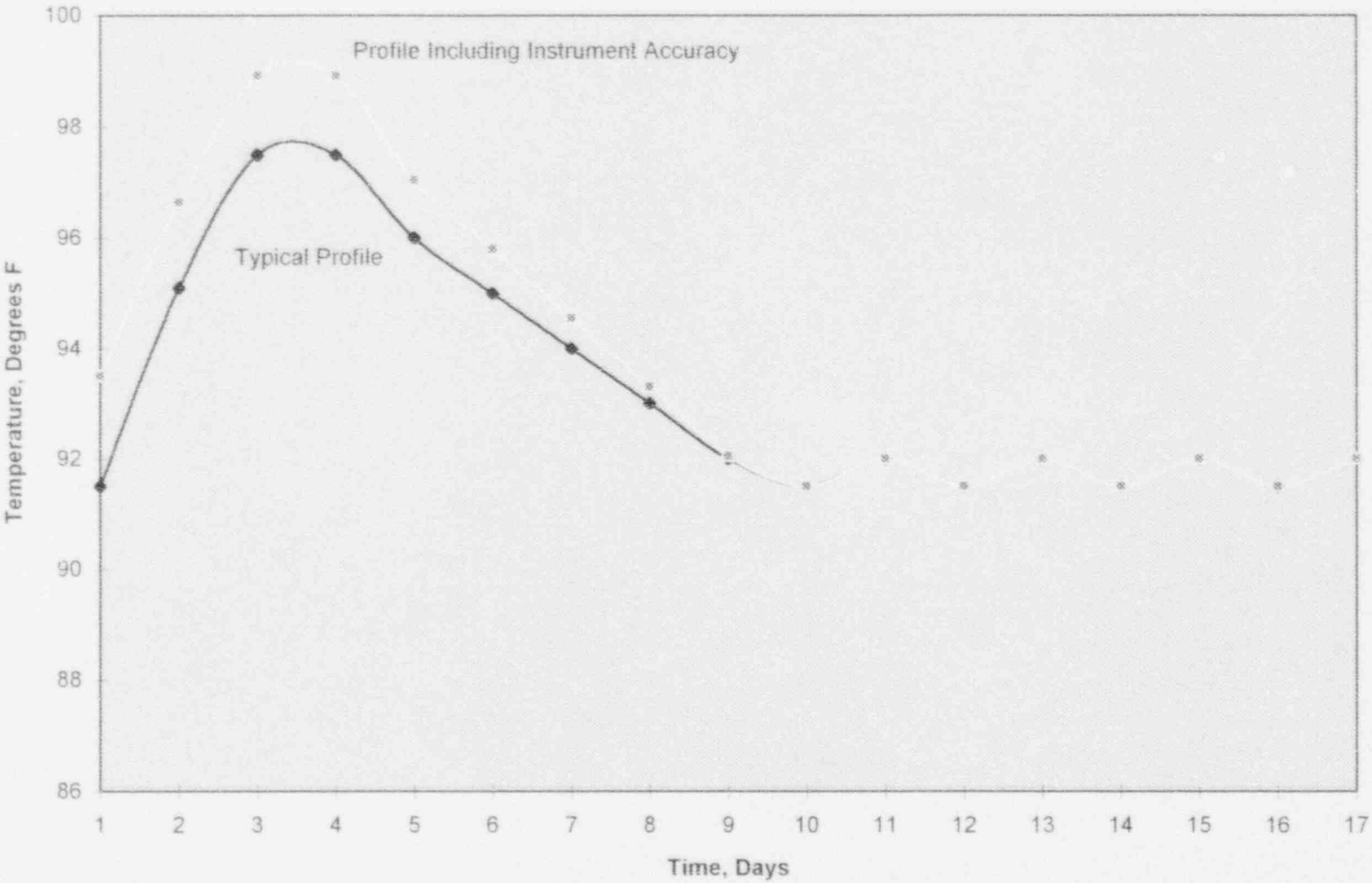
<u>SNSWP Level</u>
The installed SNSWP Operator Aid Computer (OAC) level instrumentation has an accuracy of approximately ± 0.16 ft (CNC-1210.04-00-0069). In a detailed mechanistic pond analysis, the effect of SNSWP level instrument inaccuracy would be to establish an offset in the starting point of both the initial inventory and available heat transfer area. However, the Catawba SNSWP analysis is performed using a simple but conservative analytical model which performs the analysis at a SNSWP level significantly below the Technical Specification surveillance value. This is consistent with the methodology discussed above and conservatively bounds the effect of SNSWP level instrumentation inaccuracy.

<u>SNSWP Temperature</u>
The installed SNSWP OAC temperature instrumentation has an accuracy of approximately ± 2.13°F (CNC-1210.04-00-0067). The effect of SNSWP temperature instrument inaccuracy could be a non-conservative increase in the initial energy content of the SNSWP. The SNSWP analysis methodology conservatively assumes that the SNSWP is a homogeneous mixture of water at the same temperature as measured by the SNSWP temperature instrumentation at 568 ft msl rather than a temperature-stratified pond with colder water below the temperature monitoring instrument. This analysis assumption has the effect of conservatively increasing the initial energy content of the SNSWP for the analysis and would offset, to some degree, if it were rigorously analyzed, the possible non-conservative effect of the SNSWP temperature instrument inaccuracy. Ignoring the conservatism of the analysis methodology, the effect of increasing the initial SNSWP temperature would be a short-term effect because this additional energy is removed from the SNSWP and rejected to the environment prior to the end of the thirty-day SNSWP analysis. This effect can be seen in Figure 1 where the post-accident "Profile Including Instrument Accuracy" eventually decreases back down to the point where it matches the "Typical Profile".

Standby Nuclear Service Water Pond level and temperature are inputs into the analysis which determines the post-accident temperature of the SNSWP. Post-accident SNSWP temperature is an input for the following analysis:

1. <u>The containment accident analysis (i.e., peak accident pressure).</u> The effects of SNSWP temperature and level instrument inaccuracy on the containment analysis have been reviewed and determined to be of no safety significance. This is based on the

FIGURE 1. POST-ACCIDENT INSTRUMENT ACCURACY PROFILE FOR SNSWP TEMPERATURE



considerable margin present in the containment design. The following is a summary of the available margins and conservatism in the containment analysis:

- The containment analysis starts at 92°F instead of the SNSWP temperature surveillance limit of 91.5°F.

- The current analyzed containment pressure is 14.05 psig.

- The Technical Specification Surveillance value for containment pressure testing and analytical limit for containment pressure analysis is 14.68 psig.

- The containment design pressure is 15 psig. This is documented in FSAR Section 6.2.1.1.1 and in Catawba Technical Specification 5.2.2.

- The structural acceptance test for the containment vessel is performed at 110% to 115% of design pressure (16 psig to 17.25 psig).

- The containment ultimate capacity is 72 psig. This is the result of an ultimate capacity analysis documented in FSAR Section 3.8.2.5.3.

The effects of instrument inaccuracy associated with the SNSWP level and temperature on the containment analysis was determined to have an impact of less than 0.14 psi. This demonstrates that instrument inaccuracy has an insignificant impact on the containment design function and may appropriately be discounted in the analysis and the surveillance procedures.

2. Post-accident cooldown of containment. The SNSWP serves as the ultimate heat sink for removing heat from the containment. Post-accident cooldown of the containment is required to assure that equipment required for long term cooling of the core is capable of performing its intended function. Equipment located inside of containment has been qualified for a one-time accident temperature excursion for a minimum of ten days. Most equipment was LOCA tested for thirty days. The currently-analyzed accident temperature excursion lasts approximately eight days. The effects of the SNSWP temperature instrument inaccuracy on the post-accident cooldown on containment are insignificant. Even if the initial SNSWP temperature instrument inaccuracy was applied conservatively as bias which lasted the entire duration of the accident, containment would still cool down within the eight days as currently stated in the analysis. This still allows considerable margin to the ten days the equipment is qualified for. (Reference: The Catawba Environmental Qualification Criteria Manual Figure 6.0-4.)

3. Equipment qualification for equipment outside of containment, for which the SNSWP is the ultimate heat sink. The SNSWP serves as the ultimate heat sink for equipment outside of containment which also has equipment qualification requirements. This equipment would include the nuclear service water pump motors and motors cooled by the component cooling system. The equipment qualifications for this equipment were very conservatively established. Qualifications for equipment cooled by the component cooling water system (KC), for example, were determined on the basis of running at full load and maximum normal KC system temperature of 100°F rather than the normal part load operation at 90 to 95°F. This has the effect of steadily building margin in the qualified life of the equipment. It also includes an allowance for a fast unit shutdown every year which causes a short-term KC system temperature excursion to 130°F which has never happened. A short-term increase in the heat rejection temperature of a

magnitude equal to the inaccuracy of the SNSWP temperature instrumentation would have an insignificant impact on the qualified life of this equipment.

4. <u>Control Room Chiller operation.</u> The Standby Nuclear Service Water Pond serves as the ultimate heat sink for the control room ventilation and chilled water systems. In accordance with Technical Specifications, the control room is verified to be less than 90°F. Control room temperature is controlled for equipment qualification and operator comfort concerns. The control room temperature is routinely maintained at approximately 74°F which allows for considerable margin to equipment qualification requirements (typically based on 104°F ambient). This also has the effect of steadily building margin in the qualified life of equipment in the control room. Under the current control room chiller analysis, control room temperature would be maintained at approximately 80°F under the most severe heat loading assumptions. A short-term temperature excursion which increased this temperature by a magnitude equal to the SNSWP temperature instrument inaccuracy would have an insignificant effect on the qualified life of equipment located in the control room. Additionally, by the time this temperature was reached, at least twelve hours into an accident, most control room equipment would have performed its safety function with post-accident monitoring being the primary function remaining.

In summary, the overall impact of SNSWP temperature and level instrumentation inaccuracy is insignificant on safety, and therefore instrument inaccuracy has been appropriately accounted for. Catawba elected to not specifically apply an instrument inaccuracy for these items in the past, and, consistent with Catawba's overall philosophy and site directives on instrument inaccuracy, the surveillance procedure did not include an allowance for instrument inaccuracy. This approach is consistent with Westinghouse philosophy and with applicable regulations. Combined with the fact that the surveillances were actually performed in accordance with Technical Specifications, Catawba denies the violation.

An overall review of the application of instrument inaccuracy at Catawba has been conducted. The results of this review found that in general:

- An allowance for instrument inaccuracy has been included in Chapter 15 accident analyses which deal with core protection, such as ECCS analyses, or it has been determined to be insignificant.

- An allowance for instrument inaccuracy has not been included in the Chapter 6, Containment Analysis (see the earlier discussion of margin to safety in the containment analysis).

- An allowance for instrument inaccuracy has been included in all safety-related instrumentation setpoint determinations (note that the SNSWP level and temperature instrumentation are not safety-related).

This is consistent with the overall philosophy of applying instrumentation inaccuracy allowances only in those cases where it has an impact on safety.

Duke Power Company has recognized that, in the past, issues involving instrument inaccuracy overall have not been clearly defined, documented, and controlled, and training has not been provided to the appropriate personnel. Duke Power Company is taking corrective actions to address this deficiency. These corrective actions were initiated prior to the Service Water System Operational Performance Inspection. The corrective actions include actions undertaken by the

Electrical Instrumentation and Control (I&C) Best Team and a Nuclear Station Testing Quality Improvement Team. The Electrical I&C Best Team was tasked to develop a consistent company-wide approach to the evaluation and application of instrumentation inaccuracy to setpoint determination. A final draft of the document generated as a result of this effort is being circulated for final comments at this time and should be approved and issued as an addition to the Engineering Documents Manual in the near future. The Nuclear Station Testing Quality Improvement Team charter includes the establishment of a consistent testing methodology at Duke Power Company's nuclear stations, one aspect of which includes the control and application of instrument error in nuclear station testing which includes Technical Specification surveillance procedures. The results of this Quality Improvement Team will be issued as a rewrite of Nuclear System Directive 408, "Post-Maintenance Testing," and should be issued for review as a first draft in November of 1994. The results of these efforts should provide: the establishment of clear criteria for determining if a given variable requires inclusion of instrument inaccuracy, the establishment of a consistent methodology for determining the instrumentation inaccuracy for a given process variable, the establishment of a process which provides assurance that the instrumentation inaccuracy is appropriately applied to the process variable either in the analysis or, in rare cases, the surveillance procedure, and documentation of Duke Power Company's instrument inaccuracy position to allow for consistent personnel training in the future.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-15

Notice of Violation

F. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires in part, that measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

Contrary to the above:

1. As of August 1, 1994 the test acceptance criteria drawing CNTC-2573-KC-H002-02 for the number of allowable plugged tubes in the 2B component cooling water heat exchanger was 600 tubes and the heat exchanger was in service even though problem identification report 2-C94-0680 stated the number of allowable plugged heat exchanger tubes would be revised to 675 before returning the heat exchanger to service.

2. In November, 1993, when the differential pressure to the control room chiller was determined to be less than design requirements a problem identification report was not written or corrective action taken to evaluate the chiller for operability.

3. As of August 1, 1994, safety related service water pump 1A was not being operated for at least three hours per week even though problem identification report 1-C94-0260 stated the pump would be so operated.

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

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-15

RESPONSE: (General)

1. **Reason for Violation**

Although each of the examples is slightly different relative to the surrounding circumstances (note that Catawba is denying violation example F2), from an overall perspective there was a lack of understanding of management's expectations concerning initiating actions in response to identified problems and concerning implementing previously identified corrective actions. Therefore, this violation is attributed to management deficiency.

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

Catawba has developed a training package concerning the Problem Investigation Process (PIP). This package will enhance the understanding of plant personnel relative to the initiation of corrective actions in response to identified problems. Engineering has been trained using this new package. Plans are being formulated to train all site personnel in preparation for the Unit 1 end-of-cycle 8 refueling outage.

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

Catawba management remains committed to improving the process by which corrective actions are both identified and implemented in response to known problems. Catawba continues to conduct quarterly reviews of corrective action effectiveness. These reviews evaluate whether corrective actions taken in response to identified problems are appropriate to address both the initiating problem and the identified root cause. In addition, Catawba utilizes industry experts to assist in the corrective action effectiveness reviews. Finally, periodic reviews of control room logs are conducted to determine whether events as documented in the logs meet the criteria for PIP initiation.

4. **Date of Full Compliance**

Catawba Nuclear Station is in full compliance.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-15

RESPONSE: (Example F1)

1. **Reason for Violation**

The Test Acceptance Criteria (TAC) drawing CNTC-2573-KC-H002-02 was not revised prior to the component cooling (KC) heat exchanger 2B being placed back in service. PIP 2-C94-0680 identified that the tubes in the heat exchanger that had to be plugged would exceed the plugging limit, and stated that the number of allowable plugged tubes would be revised from 600 to 675 on the TAC sheet prior to placing the heat exchanger back in service.

This example is attributed to a design change not properly coordinated with design change implementation.

The requirement to update the TAC sheet was identified in the PIP communicated to the service water systems engineer. No corrective action was assigned in the PIP to track the revision of the TAC sheet. As a result, the TAC sheet calculations were completed but not checked and signed prior to the heat exchanger being placed back in service.

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

The plugged tube limit on the TAC sheet has been revised. Additionally, the entire process which governs changes to KC heat exchanger tube plugging was flowcharted and reviewed. Programmatic changes have been initiated to ensure the TAC sheet tube plugging limit is checked whenever heat exchanger tubes are plugged. Current policies were discussed and emphasized to ensure that a PIP is initiated and corrective action is assigned if the plugging limit will be exceeded, and to initiate a minor modification to document the change to the heat exchanger and TAC sheet if necessary.

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

Specific programmatic changes have been initiated to add a step in the generic maintenance procedure on heat exchanger eddy current testing and the generic maintenance procedure on heat exchanger tube plugging to ensure the plugged tube limit is checked on the TAC sheet prior to placing the heat exchanger back in service. All procedure changes will be complete prior to the start of the Catawba 1 end-of-cycle 8 refueling outage, scheduled to begin on February 3, 1995.

4. **Date of Full Compliance**

Since the TAC sheet has been revised, Catawba Nuclear Station is in full compliance.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-15

RESPONSE: (Example F2)

1. **Basis for Denying Violation**

This violation example involves failure to initiate a problem identification report or take corrective action when the differential pressure to a control room chiller was found to be less than design requirements. Catawba is denying this violation example, since it is maintained that the condition adverse to quality referred to was indeed properly identified, corrective actions were promptly initiated, and the procedures and programs in place at the time of this incident were properly implemented with respect to the determination of chiller operability.

In late September 1993, it was noted by Engineering that the data from the previous Train A and Train B RN flow balances conducted on September 16 and 23, 1993, respectively, were significantly lower than data available from the manufacturer. Although no specific acceptance criteria were provided for this parameter, the inconsistency between actual data versus vendor data was determined to warrant further investigation.

When this discrepancy was first identified in late September, actions were initiated to obtain sufficient data to adequately address chiller operability. Both chillers were performing adequately at the time under normal heat loads with significant additional valve travel available. Catawba believes that the following corrective actions were both prompt and effective:
1) cleaning the RN supply and return piping to the chillers, 2) developing an extensive testing and monitoring program to trend RN flow to the chillers, and 3) formulating plans for piping replacement.

Preparation for a special test was initiated on September 30, 1993 with the final test procedure approved on October 25, 1993. Tests were conducted on October 27 and 28, 1993 to determine the required flow rate and corresponding pressure drop required for the chiller to perform its design function. Final documentation of a formal operability evaluation, consistent with site procedures in place at the time, was completed on November 1, 1993.

With respect to why a PIP was not initiated at the time this problem was identified, site procedures in place at the time did not require this process to be used for operability evaluations. Site Directive 3.1.14, "Operability Determination," which was used to address operability of the control room chiller, did not require a PIP to be initiated. Based on the fact that the normal heat loads are approximately equal to accident heat loads, Engineering did not believe that there was an immediate operability concern. Therefore, a formal operability evaluation was not initiated. Problem investigation reports (PIPs) 0-C94-1123 and 0-C94-1183 have since been initiated concerning this incident. Also, a past operability evaluation has subsequently been performed and it was determined that the chiller was past operable. Therefore, no reportability concern exists.

Since this event occurred, Site Directive 3.1.14 was revised (on May 30, 1994) to require that all operability evaluations be documented using the PIP process. This revision made the operability process consistent with Nuclear System Directive 208, "Problem Investigation Process (PIP)," and ensures improved documentation and tracking of operability determinations.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION
50-413, 414/94-17-15

RESPONSE: (Example F3)

1. **Reason for Violation**

In resolving problem report 1-C94-0260, Engineering did not assign Operations a trackable corrective action with an assigned due date to ensure that the 1A RN pump was run for at least three hours per week. Communications between Engineering and Operations were not adequate and left Engineering with the impression that the administrative controls necessary to run the pump were in place. Also, the operability of the pump was not dependent upon the three-hour weekly run and the means used to notify Operations at the time were inappropriately viewed by Engineering as being adequate. This breakdown in communication is viewed as the root cause of this particular problem.

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

Operations has added the 1A RN pump to the weekly idle equipment list and it is run three hours per week minimum. This periodic run will ensure that the RN pump 1A motor is maintained in a dry condition. This is only an equipment maintenance activity and is not required for continued operability. Also, Site Directive 3.1.14, "Operability Determination", has been revised to be consistent with Nuclear System Directive 208, "Problem Investigation Process (PIP)", to clarify the method of communicating conditional operability to the Operations group. This completes the administrative tie between the two directives and clarifies when each directive should be used with the other.

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

NSD 208 will be updated to ensure that any immediate corrective action documented in a problem investigation report, which is not complete at the resolution stage of the report, will have a corrective action with a due date assigned to the responsible group. This corrective action will be completed by December 31, 1994.

4. **Date of Full Compliance**

Catawba Nuclear Station is in full compliance.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
REPLY TO NOTICE OF VIOLATION

Discussion of Plans for Service Water System Chemical Treatment

This supplemental discussion is being provided in response to a request made by the NRC in the cover letter transmitting the subject inspection report. In the cover letter, the NRC indicated a special concern over the lack of service water system chemical treatment at Catawba. The NRC specifically requested that Catawba submit a description of actions to improve service water system water quality or to initiate chemical treatment.

Background
Catawba believes that chemical control of the nuclear service water system could potentially yield positive results in mitigating the effects of siltation, corrosion, and macrofouling which are caused by the site's raw water. Previous studies of raw water at the site have investigated the corrosive effect of the water on various materials and welds on the materials, as well as the effects of flow on the corrosion rates of these materials. In addition, Catawba has performed a study to evaluate CT-1 as a chemical control for macrofouling. The use of CT-1 resulted in unacceptable foaming problems and questionable control effectiveness. Also, it was determined that the detoxification agent itself was toxic to the environment. The State of South Carolina has imposed toxicity test protocols unique in the United States, but which carry constraints statewide, under state and federal environmental protection acts. Nevertheless, Catawba's plan is to evaluate additional options for chemical control and to develop a contingent control strategy for potential future infestation of Zebra Mussels. By November of 1995, Catawba will determine which chemicals are most effective in mitigating siltation, corrosion, and macrofouling for the service water system. Catawba will initiate field studies to corroborate positive results from the addition of chemicals prior to implementing a full-scale chemical addition program. Due to the required exposure time of the corrosion test coupons (specimens), Catawba anticipates that it will take approximately twelve months to obtain results upon which to base a chemical addition program. Permit licensing and capital additions for equipment will be pursued for chemical addition of those chemicals which provide significant short and/or long-term benefits. In the event that Zebra Mussel infestation is detected in the lake, Catawba will take immediate action to inject a biocide. Lake surveys for the mussels, which have been in progress since 1992, will provide the data upon which to base this action.

Current Plans
As discussed above, Catawba had expected CT-1 to serve as the chemical control product, but it was unacceptable. Catawba hopes to qualify another chemical, either sodium hypochlorite or a mixture of sodium hypochlorite and bromine, by November of 1995 and be in a position to begin plant modifications required for implementation. To compensate for this delay, Catawba will continue to aggressively monitor the health of the system, as well as monitor the lake for both Zebra Mussel infestation and any significant change in the infestation of Asiatic Clams.