

September 22, 2004

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

Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Technical Specifications and Bases
Amendment
Technical Specification and Bases 3.6.10
Annulus Ventilation System (AVS)
Technical Specification and Bases 3.6.16
Reactor Building
Technical Specification Bases 3.7.10
Control Room Area Ventilation System (CRAVS)
Technical Specification Bases 3.7.12
Auxiliary Building Filtered Ventilation Exhaust
System (ABFVES)
Technical Specification Bases 3.7.13
Fuel Handling Ventilation Exhaust System (FHVES)
Technical Specification and Bases 3.9.3
Containment Penetrations
Technical Specification 5.5.11
Ventilation Filter Testing Program (VFTP)
TAC Numbers MB7014 and MB7015

- References:
1. Letters from Duke Energy Corporation to NRC, dated November 25, 2002, November 13, 2003, and December 16, 2003
 2. Letter from NRC to Duke Energy Corporation, dated May 25, 2004

In Reference 2, the NRC provided a Request for Additional Information (RAI) concerning the Catawba license amendment request submittal transmitted via the Reference 1 correspondence. The purpose of this letter is to submit a partial response to the NRC RAI. This letter contains responses to the RAI questions on filter test criteria. In

A001

U.S. Nuclear Regulatory Commission
Page 2
September 22, 2004

addition, this letter contains responses to several chemistry related questions provided by one of your technical reviewers in a conference call. Duke Energy Corporation anticipates submitting a response to the RAI question on the use of MOX fuel by December 8, 2004.

Duke Energy Corporation has determined that the original No Significant Hazards Consideration Analysis and Environmental Analysis contained in the license amendment request submittal of November 25, 2002 are unchanged as a result of this RAI response.

Pursuant to 10 CFR 50.91, a copy of this letter is being sent to the appropriate State of South Carolina official.

Inquiries on this matter should be directed to L.J. Rudy at (803) 831-3084.

Very truly yours,

A handwritten signature in black ink, appearing to read 'D.M. Jamil', with a stylized flourish at the end.

D.M. Jamil

Attachment

U.S. Nuclear Regulatory Commission
Page 4
September 22, 2004

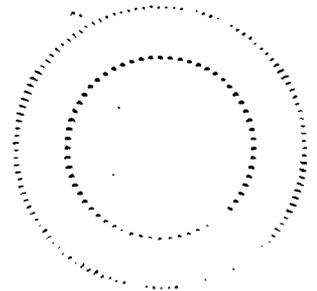
xc (with attachment):

W.D. Travers
U.S. Nuclear Regulatory Commission
Regional Administrator, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

E.F. Guthrie
Senior Resident Inspector (CNS)
U.S. Nuclear Regulatory Commission
Catawba Nuclear Station

S.E. Peters (addressee only)
NRC Project Manager (CNS)
U.S. Nuclear Regulatory Commission
Mail Stop O-8 G9
Washington, D.C. 20555-0001

H.J. Porter, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental Control
2600 Bull St.
Columbia, SC 29201



ATTACHMENT

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

May 25, 2004 NRC RAIs on Filter Test Criteria

1. In your application, you proposed changing the criteria for in-place testing of the Catawba, Unit 2 High Efficiency Particulate Air (HEPA) filters (Annulus Ventilation, Auxiliary Building Exhaust, and Fuel Building Ventilation) from 0.05% penetration to 1% penetration and re-labeling the test criteria "penetration and system bypass." TS 5.5.11a states that this value will be the penetration and system bypass "when tested in accordance with Regulatory Guide (RG) 1.52, Revision 2 and ANSI N510-1980 at the flow rate specified..." The purpose of the test is to verify the integrity of the HEPA filter assembly to assure that there are no gaps or pathways that allow the effluent to circumvent the filter media either around the media or through the media. RG 1.52, Revision 2 specifies a criteria of 0.05% for bypass flow during in place testing which has been attainable by Catawba in the past and by most other licensees. Please provide a justification for non-compliance with the NRC staff position expressed in the regulatory guide. A discussion of the significance of the in-place leak test is provided in ANSI N510-1975 Appendix B. Section B-2 pertains to HEPA filters.

Response: As stated in Attachment 1, Page 21 of the November 13, 2003 submittal, Catawba is requesting a consistent application of the 1% in-place penetration and system bypass leakage design basis limit for the Unit 2 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES), Annulus Ventilation System (AVS), and Fuel Handling Ventilation Exhaust System (FHVES) HEPA filters. The 1% in-place HEPA filter penetration and system bypass leakage value is the technical specifications limit for the Catawba Unit 1 system filtration units, as well as the McGuire Nuclear Station Unit 1 and Unit 2 corresponding ventilation system filtration units. The bases for the HEPA filter penetration and bypass leakage limit is to limit the release of radioactive particulates after a design basis event. With a technical specification 1% HEPA penetration and system bypass leakage limit, a logical nexus would be drawn between Catawba's surveillance tests and the dose analyses per 10 CFR 50.36.

Overall, the impact of a 1% HEPA filter penetration and system bypass leakage on the dose analyses is small relative to all other design inputs and assumptions.

As stated in Attachment 3, Page 25 of the November 25, 2002 license amendment request submittal, no HEPA filtration credit is assumed for the ABFVES in the current design basis Loss of Coolant Accident (LOCA) dose analysis because

particulate iodine is not released through the Emergency Core Cooling Systems (ECCS). In Attachment 3, Page 31 of the November 25, 2002 submittal, Catawba states that credit is assumed for HEPA filtration in the Fuel Handling Accident (FHA) dose analysis involving recently irradiated fuel. However, Catawba does not move recently irradiated fuel and no credit for FHVES HEPA filtration is assumed in the FHA dose analysis involving non-recently irradiated fuel.

The Catawba and McGuire Nuclear Stations are licensed to Regulatory Guide 1.52, Revision 2, and perform HEPA filter in-place penetration and bypass leakage testing in accordance with ANSI N510-1975 and 1980. The Catawba and McGuire technical specifications have always referred to the subject test as an "in-place penetration and system bypass leakage test."

ANSI N510-1975, Section B-2 and ANSI N510-1980, Section A-2 were reviewed and Catawba does not disagree with the subject discussions. Catawba does not consider the HEPA filter in-place penetration and system bypass leakage test to be an efficiency test. Penetration and system bypass as defined by ANSI N510 is the concentration of the DOP aerosol gas which leaks through and/or around the HEPA filter bank. If HEPA filtration is credited within the dose analyses, all penetration and system bypass leakage is treated as an unfiltered release within the dose analyses.

The HEPA filtration unit efficiency (95%) used within the dose analyses assumes that 5% of the radioactive particulates penetrate or bypass the HEPA filter bank. The 5% of radioactive particulates that penetrates and/or bypasses the HEPA filters is conservatively treated as an unfiltered release within the dose analyses. As shown in Attachment 1 of the November 25, 2002 submittal, the historical ABFVES, AVS, and FHVES in-place penetration and system bypass leakage test results infer that the particle removal efficiency of all the subject HEPA filtration systems is greater than 99.97%. With HEPA filters designed for a 0.3 micrometer DOP aerosol particle removal efficiency of 99.97%, a 1% penetration and bypass leakage limit would not impact the HEPA filtration efficiency margin historically utilized within the dose analyses. The efficiency of these HEPA filtration systems would be higher for larger particles.

Therefore, as stated in Attachment 1, Page 25 of the November 13, 2003 submittal, changing the in-place penetration and system bypass leakage to 1% for the Unit 2 ABFVES, AVS, and FHVES HEPA filters does not create any safety concerns. The 1% HEPA penetration and system bypass leakage has always been the limiting operating condition for the Catawba Unit 1 and McGuire Unit 1 and 2 corresponding

systems. The 95% HEPA filtration efficiency has always been the design basis input for the dose analyses. The 0.05% technical specification HEPA penetration and system bypass leakage limit imposes an unnecessary restrictive margin for these filtration systems without any significant safety improvement.

2. *The NRC staff has a similar question with respect to the carbon adsorber filters. You proposed changing the criteria for in-place testing the Catawba, Unit 2 carbon adsorber filters (Annulus Ventilation, Auxiliary Building Exhaust, and Fuel Building Ventilation) from 0.05% penetration to 1% penetration and re-labeling the test criteria "penetration and system bypass." TS 5.5.11b states that this value will be the penetration and system bypass "when tested in accordance with Regulatory Guide 1.52, Revision 2 and ANSI N510-1980 at the flow rate specified..." The purpose of the test is to verify the integrity of the carbon adsorber filter assembly to assure that there are no gaps or pathways that allow the effluent to circumvent the filter media either around the media or through the media. The RG specifies a criteria of 0.05% for bypass flow during in place testing which has been attainable by Catawba and most other licensees in the past. Please provide a justification for non-compliance with the NRC staff position expressed in the RG 1.52, Revision 2. A discussion of the significance of in-place leak test is provided in ANSI N510-1975 Appendix B. Section B-3 pertains to carbon adsorbers.*

Response: As stated in Attachment 1, Page 21 of the November 13, 2003 submittal, Catawba is requesting a consistent application of the 1% in-place penetration and system bypass leakage design basis limit for the Unit 2 ABFVES, AVS, and FHVES carbon bed filters. The 1% in-place carbon filter penetration and system bypass leakage value is the technical specifications limit for the Catawba Unit 1 system filtration units, as well as the McGuire Nuclear Station Unit 1 and Unit 2 corresponding ventilation system filtration units. The bases for the carbon filter penetration and bypass leakage limit is to limit the release of elemental iodine and organic iodide after a design basis event. With a technical specification 1% carbon filter penetration and system bypass leakage limit, a logical nexus would be drawn between Catawba's surveillance tests and the dose analyses per 10 CFR 50.36.

Overall, the impact of a 1% carbon filter penetration and system bypass leakage on the dose analyses is small relative to all other design inputs and assumptions.

As stated in Attachment 3, Page 30 of the November 25, 2002 license amendment request submittal, Catawba states that credit is assumed for carbon filtration in the FHA dose analysis involving recently irradiated fuel. However, Catawba does not move recently irradiated fuel and no credit for the FHVES carbon filtration is assumed in the FHA dose analysis involving non-recently irradiated fuel.

The Catawba and McGuire Nuclear Stations are licensed to Regulatory Guide 1.52, Revision 2, and perform carbon bed filter in-place penetration and bypass leakage testing in accordance with ANSI N510-1975 and 1980. The Catawba and McGuire technical specifications have always referred to the subject test as an "in-place penetration and system bypass leakage test."

ANSI N510-1975, Section B-3 and ANSI N510-1980, Section A-3 were reviewed and Catawba does not disagree with the subject discussions. Catawba does not consider the carbon filter bed in-place penetration and system bypass leakage test to be an efficiency test. Penetration and system bypass as defined by ANSI N510 is the concentration of the refrigerant tracer gas, such as R-11, which leaks through and/or around the carbon filter bed.

As stated in ANSI N510-1975, Section B-3 and ANSI N510-1980, Section A-3, adsorption is time dependent and therefore instantaneous contaminant removal efficiency is meaningless. Therefore, nuclear safety related carbon filter media is tested throughout the industry using the ASTM 3803-1989 low temperature and high humidity pre-equilibrium tests to determine the percent penetration and efficiency. In these laboratory tests, the methyl iodide competes with water vapor and other contaminants for adsorption sites on the carbon filter media. The ASTM 3803-1989 standard test is recognized as the single best method for determining carbon filter media penetration and efficiency. The Catawba and McGuire Nuclear Stations utilize the ASTM 3803-1989 test method to determine the carbon filter media percent penetration and efficiency.

In the carbon filter in-place penetration and system bypass tests, Refrigerant 11 (R-11) competes with water vapor and other contaminants for adsorption sites on the carbon filter media. New carbon is known to adsorb R-11 quite effectively and release its R-11 very slowly, while used/wet carbon tends to adsorb R-11 poorly and release its adsorbed refrigerant quite rapidly. This effect is due to adsorbed organic material and water blocking the carbon's adsorption sites. As the carbon filter media ages and is exposed to contaminants and high humidity air, the R-11 adsorptive properties are reduced. The collection of contaminants and

moisture within the carbon filter media reduces the available internal surface area for the adsorption of R-11. Thus, the carbon filter media R-11 tracer gas desorption rate increases and penetration through the carbon filter media increases (Ref. 4, 5).

The historical ABFVES, AVS, and FHVES carbon filters in-place penetration and system bypass test results are shown below for the past six years. For the carbon filter test results with penetration and system bypass greater than 0.05%, the ASTM 3803-1989 laboratory test results are also included. The ASTM 3803-1989 laboratory penetration values were well below the technical specification 4% limit while the in-place penetration and system bypass were greater than 0.05%. For the Unit 2 filtration units, the carbon filter media was replaced and the integrity of the filter bed assemblies was not a factor relative to the penetration and bypass leakage.

System/Filtration Unit	Date	HEPA Leakage (%)	Carbon Leakage (%)	ASTM 3803-1989 Carbon Laboratory Penetration (%)
ABFVES/1A	10/29/98	0.0060	N/A	
	12/2/98	N/A	0.0000	
	2/23/00	0.0040	N/A	
	1/17/01	0.0120	0.0000	
	7/3/01	0.0008	0.0000	
	8/29/02	0.0100	N/A	
	5/7/03	0.0020	0.2153	0.28
	1/14/04	0.0050	0.1252	0.82
ABFVES/1B	7/30/98	0.0020	N/A	
	12/9/98	0.0020	0.0000	
	12/15/99	0.0040	N/A	
	5/25/00	0.0070	0.0000	
	1/31/01	N/A	0.0000	
	7/20/01	0.0140	0.0000	
	5/21/03	0.0100	0.2260	0.35
	2/1/04	0.0120	0.2833	0.27
AVS/1A	10/20/98	0.0014	0.0080	
	8/31/00	0.0010	0.0002	
	4/17/02	0.0030	0.0013	
	1/20/04	0.0020	0.0030	
AVS/1B	7/1/99	0.0025	0.0000	

	5/3/01	0.0036	0.0000	
	9/19/02	0.0005	0.0000	
	2/5/04	0.0016	0.0009	
FHVES/1A1	4/15/99	0.0040	0.0000	
	2/27/01	0.0070	0.0000	
	6/6/02	0.0040	0.0000	
	3/19/04	0.0050	0.1065	0.97
FHVES/1A2	2/18/99	N/A	0.0015	
	4/15/99	0.0050	0.0000	
	2/27/01	0.0080	0.0000	
	6/6/02	0.0050	0.0000	
	3/19/04	0.0015	0.0091	
FHVES/1B1	5/26/99	0.0048	0.0000	
	3/27/01	0.0043	0.0000	
	3/3/04	0.0008	0.0000	
FHVES/1B2	5/26/99	0.0021	0.0000	
	3/27/01	0.0043	0.0013	
	3/3/04	0.0012	0.0000	
ABFVES/2A	8/15/98	0.0020	N/A	
	10/28/99	0.0010	0.0000	
	12/29/99	0.0100	N/A	
	2/21/01	0.0060	N/A	
	9/6/01	0.0014	0.0960	0.10
	9/15/01	N/A	0.0000	
	10/31/01	0.0028	N/A	
	6/11/03	0.0170	0.3314	0.11
	6/14/03	N/A	0.0313	
	4/8/04	0.0040	0.0324	
ABFVES/2B	8/26/98	0.0160	N/A	
	8/23/99	0.0025	0.3700	0.18
	8/25/99	N/A	0.3200	0.18
	8/27/99	N/A	0.0000	
	1/13/00	0.0040	N/A	
	3/7/01	0.0053	0.0000	
	5/30/01	0.0060	N/A	
	7/24/02	0.0100	0.0000	
	6/25/03	0.0120	N/A	
	3/22/04	0.0080	0.0000	
AVS/2A	11/30/98	0.0030	0.0000	
	7/13/00	0.0026	0.0014	
	11/27/01	0.0032	0.0000	
	7/31/03	0.0020	0.0000	

AVS/2B	12/7/98	0.0034	0.0000	
	3/1/00	0.0200	N/A	
	6/29/00	0.0030	0.0000	
	1/31/02	0.0045	0.0053	
	7/16/03	0.0040	0.0000	
FHVES/2A1	4/16/98	0.0140	0.0000	
	10/7/99	0.0040	0.0000	
	3/20/01	0.0050	0.0000	
	8/8/02	0.0042	0.0000	
	3/17/04	0.0015	0.0963	0.92
	7/9/04	N/A	0.0406	
FHVES/2A2	4/16/98	0.0054	0.0030	
	10/7/99	0.0064	0.0000	
	3/20/01	0.0100	0.0038	
	8/8/02	0.0040	0.0027	
	3/17/04	0.0065	0.0248	
FHVES/2B1	11/12/98	0.0080	0.0000	
	6/21/00	0.0085	0.0000	
	12/6/01	0.0080	0.0330	
	4/25/03	0.0060	0.0000	
FHVES/2B2	11/12/98	0.0034	0.0004	
	6/21/00	0.0035	0.0000	
	12/6/01	0.0050	0.0000	
	4/25/03	0.0040	0.0000	

The ABFVES operates in the continuous filtration mode to eliminate a single failure concern associated with operating in the filter bypass mode and thus increases reliability and availability of the system. The majority of the airflow is unheated and the carbon filter media is continuously exposed to contaminants and water vapor. Since the ABFVES operates in a continuous filtration mode, the collection of contaminants and water vapor within the carbon filter media can affect the retention of R-11 on the carbon filter media and increase the penetration through the carbon filter media during field tests. The 0.05% in-place penetration and system bypass leakage limit then results in premature carbon filter media replacement even though the samples meet the ASTM 3803-1989 test acceptance criteria. The inability to consistently meet the 0.05% carbon filter in-place penetration and system bypass leakage criterion is a desorption issue and not a leakage issue.

Increasing the carbon filter in-place penetration and system bypass leakage technical specification from 0.05% to 1% will reduce the economic impact to the station without reducing the margins associated with radiological offsite and onsite

releases after a design basis event. The carbon bed filtration unit efficiency used within the dose analyses assumes that 5% of the radioactive particulate and elemental iodine penetrates or bypasses the carbon filter bed and 20% of the radioactive organic iodide penetrates or bypasses the carbon filter bed. The radioactive iodine and iodide that penetrates and/or bypasses the carbon filters is conservatively treated as an unfiltered release within the dose analyses. These elemental iodine and organic iodide carbon filtration efficiencies have historically been acceptable to the NRC Staff (Ref. 3).

Therefore, as stated in Attachment 1, Page 25 of the November 13, 2003 submittal, changing the in-place penetration and system bypass leakage to 1% for the Unit 2 ABFVES, AVS, and FHVES carbon bed filters does not create any safety concerns. With carbon samples tested per the ultra conservative ASTM 3803-1989 low temperature and high humidity pre-equilibrium conditions, a 1% penetration and bypass leakage limit would not impact the carbon bed filtration efficiency margin historically utilized within the dose analyses. The 1% carbon penetration and system bypass leakage has always been the limiting operating condition for the Catawba Unit 1 and McGuire Unit 1 and 2 corresponding systems. The 95% carbon bed elemental iodine and 80% organic iodide filtration efficiency has always been the design basis input for the dose analyses. The 0.05% technical specification carbon filter penetration and system bypass leakage limit imposes an unnecessary restrictive margin for these filtration systems without any significant safety improvement.

3. *You also proposed to change the flow rate for the Auxiliary Building Filtered Exhaust from 30,000 cfm to 60,000 cfm and to note that this flow rate is for two fans. The NRC staff understands that each filter will be tested individually and that the flow through the filter assuming a single failure of the other train would be approximately 37,000 cfm. Since a single failure would be the potential condition during a design basis accident, please provide justification for testing the filter at a flow rate of 30,000 cfm instead of the expected flow rate of 37,000 cfm. This question applies to in-place testing of both the HEPA and carbon adsorber filters.*

Response: As stated in the November 25, 2002 submittal, Attachment 3, Page 19, Calculation CNC-1211.00-00-0123 determined the ABFVES HEPA and carbon filter in-place penetration and system bypass testing alignment. The normal operating alignment was chosen for HEPA and carbon filter in-place penetration and system bypass testing to minimize

normal radiological protection and system interaction concerns that occur when the system is operated in an abnormal alignment, as well as the impact to station operations.

During normal operation, the ABFVES operates with two trains (an exhaust train consists of one exhaust fan and one filter unit) providing the nominal design flow rate of 30,000 +/- 10% cubic feet per minute (cfm) filtered exhaust flow per train for each unit (nominal 60,000 cfm total). After any design basis event, the total penetration and system bypass (0.05% or 1%) or unfiltered leakage for a single train in operation (37,000 cfm) would be less than the bypass or unfiltered leakage for two trains in operation (60,000 cfm). Therefore, testing in the dual train alignment is conservative relative to any design basis event offsite and onsite radiological dose releases. As previously stated, the impact of any carbon filter penetration and system bypass leakage on the dose analyses is small relative to all other design inputs and assumptions.

Operation of the ABFVES in an abnormal alignment increases noble gas concentrations within the Auxiliary Building and impacts plant personnel's ability to exit the radiation control area. The abnormal alignment also creates the potential to draw steam from the Unit 1 vent stack into other ventilation systems, including carbon filters that are shut down to support this alignment. Operation of the ABFVES in an abnormal alignment impacts station operations because numerous system components must be shut down, thereby affecting plant activities such as chemistry sampling, hot machine shop work, and laundry facility work.

Calculation CNC-1211.00-00-0123 also determined that the ABFVES ASTM 3803-1989 carbon penetration and efficiency tests will be performed at a face velocity of 48 feet per minute (37,000 cfm). The 37,000 cfm single train air flow rate is only important with regard to maintaining the carbon filter methyl iodide residence time. The subject test validates the carbon filter penetration will remain below the technical specification 4% limit and thus the residence time will not be exceeded. Therefore, the ASTM 3803-1989 carbon filter efficiency tests verify that the performance assumptions utilized within the dose analyses are maintained. For the carbon filter in-place penetration and system bypass tests, the single train alignment would not validate the residence time, since R-11 is used as the challenge gas. Operation of the ABFVES at the higher air flow rate would not adversely impact the integrity of the HEPA or carbon filters, thereby creating additional penetration and system bypass leakage.

Since only one of the two filter trains is required to initially provide the 100% (6,500 cfm) exhaust flow from the ECCS pump rooms per unit after a design basis event, the ABFVES HEPA and carbon filter in-place penetration and system bypass leakage normal testing alignment, as well as the carbon filter laboratory penetration and efficiency testing is conservative and supports the filter performance assumptions within all design basis event offsite and onsite dose analyses.

References:

- 1) Regulatory Guide 1.52, Revision 2, "Design, Testing, And Maintenance Criteria For Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration And Adsorption Units of Light-Water-Cooled Nuclear Power Plants".
- 2) ANSI/ASME N510-1975 (1980), "Testing of Nuclear Air-Cleaning Systems".
- 3) United States Nuclear Regulatory Commission, "Issuance of Amendment No. 90 to Facility Operating License NPF-35 and Amendment No. 84 to Facility Operating License NPF-52 - Catawba Nuclear Station, Units 1 and 2 (TACS 80122/80123)", dated August 23, 1991.
- 4) "Replacement Tracer Agents For The In-Place Leak Testing of Adsorbers In NATS", by J.R. Pearson, K.M. Fleming, and J.R. Hunt of NCS Corporation, Columbus, Ohio and P.L. Lagus of Lagus Applied Technology, Inc., San Diego, California.
- 5) "Refrigerant Loading Effects on Carbon During Bypass Leakage Testing", by William P. McDonald and Dennis G. Adams of the Commonwealth Edison Company and John R. Pearson of the NCS Corporation.

Telephone NRC RAIs on Chemistry

1. Provide a description of the methodology used to calculate pH at different time intervals after a LOCA. The description should specify whether hand calculations or a computer program was used for calculating pH. If it was hand calculated, describe the methodology used. If it was a computer program, describe the code and provide the inputs to the code.

Response: The methodology used to calculate the containment sump pH following a design basis LOCA at Catawba Nuclear Station is similar to the methodology for the calculation of containment sump pH following a Maximum Hypothetical Accident at Oconee Nuclear Station (Ref. 1, 2). There are some differences; most notably the ice condenser at Catawba is modeled (Oconee has no ice condenser). Thus, a synopsis of the methods used in the calculation of post LOCA containment sump pH for Catawba is presented. The analysis demonstrates the calculated values of post LOCA sump pH are somewhat closer to the lower limit (7) than they are to the upper limit (9.5). In fact, the calculated values of post LOCA sump pH cannot exceed 9.3 as the vast majority of the alkaline material is the ice in the ice condenser with sodium tetraborate with a pH of 9.3. For this reason, assumptions are made and initial and boundary conditions are set to ensure the calculation of lower bound values of post LOCA sump pH.

The calculations were performed with the use of an EXCEL spreadsheet which included the Visual Basic Program PHSC as a "macro". The spreadsheet itself was used to calculate the time dependent inventory of solutes and water (solvent), as well as the time dependent concentrations of the solutes in the containment sump. PHSC was used to calculate the time dependent pH given the concentrations of solutes in the containment sump.

The inventories of solutes and water in the containment sump were calculated by solving separate time dependent mass balance equations for the solvent and for each solute. The solutes assumed were boron in the form of boric acid and sodium tetraborate, sodium in the form of sodium tetraborate, lithium in the form of lithium hydroxide, chlorides in the form of hydrochloric acid, and nitrates in the form of nitric acid.

The inventory of water consists of contents of the Reactor Coolant System (RCS), Refueling Water Storage Tank (RWST) and cold leg accumulators, and the melted ice from the ice condenser. Based on this, the mass of water m_s at time $t > 0$ was calculated using the equation

$$m_s(t) = \int_0^t (\dot{m}_{RCS} + \dot{m}_{CLA}) d\tau + \int_0^t \dot{m}_{RWST} d\tau + m_{IC}(t)$$

where: m_{RCS} is the mass of the reactor coolant transferred to the sump,
 m_{CLA} is the mass of water in the cold leg accumulators transferred to the sump,
 m_{RWST} is the mass of water in the RWST transferred to the sump, and
 m_{IC} is the mass of melted ice from the ice condenser.

The plant technical specifications (Ref. 3) give upper and lower bounds to the inventory in the RWST and the cold leg accumulators. Since these tanks contain boric acid solutions, the upper bounds were selected in order to calculate lower bound values to the containment sump pH.

It is assumed that the contents of the RCS and cold leg accumulators are transferred to the containment sump immediately with the initiating event (time 0). This means that for any time $t > 0$,

$$\int_0^t (\dot{m}_{RCS} + \dot{m}_{CLA}) d\tau = M_{RCS} + M_{CLA} = M_{RCS} + \rho_0 V_{CLA}$$

where M_{RCS} and M_{CLA} are, respectively, the mass of the reactor coolant and the water in the (four) cold leg accumulators. Taking the density of water in the cold leg accumulators at standard conditions (ρ_0) gives $M_{CLA} = \rho_0 V_{CLA}$ with the water volume in the cold leg accumulators denoted as V_{CLA} .

The pumps of the Emergency Core Cooling System (ECCS) and Containment Spray System (CSS) transfer water from the RWST to the RCS. It is assumed that the ECCS and CSS pumps transfer water from the RWST to the containment sump; accumulation of water in the RCS and any associated delays are ignored. Maximum safeguards (both trains of the ECCS and CSS) and maximum flow rates for the ECCS and CSS pumps are taken. During the injection phase, the rate of transfer of water from the RWST then is

$$m_{RWST} = \int_0^t \dot{m}_{RWST} d\tau = \rho_0 (Q_{ECCS} + Q_{CSS})$$

where Q_{ECCS} and Q_{CSS} are, respectively, the flow rates of the ECCS and CSS pumps.

Cold leg recirculation is assumed to begin when the water in the RWST is simulated to reach the low level setpoint. At that point, only the CSS pumps are assumed to be aligned to the RWST. Then

$$m_{RWST} = \int_0^t \dot{m}_{RWST} d\tau = \rho_0 Q_{CSS} .$$

When the water in the RWST is simulated to reach the low low level setpoint, the transfer to cold leg recirculation is assumed to be complete and no pumps are aligned to the RWST.

A lower bound is taken for the mass of ice in the ice condenser (technical specification limit and an allowance for sublimation). The amount of water transferred from the ice condenser is obtained from Figure 1.

Boric acid is dissolved in the water in the RWST and cold leg accumulators. The Core Operating Limits Report (COLR) (Ref. 4) gives upper and lower limits on boron concentration in these tanks; the upper bounds were taken. The reactor coolant also may contain boric acid, depending on time into the fuel cycle or during startup. The boron concentration in the reactor coolant is set to the maximum value associated with the early phases of unit startup (refueling boron). The reactor coolant and the contents of the cold leg accumulators are assumed to be in the containment sump instantly with the initiating event (time 0). The rate of transfer of water and boric acid from the RWST is equal to the sum of the flow rates of the ECCS and CSS. The assumptions above pertaining to the inventory and transfer of the contents of the RCS, cold leg accumulators, and RWST to the containment sump are made to ensure the calculation of lower bound values of post LOCA containment sump pH. The boron concentration in the sump $[B]_s$ at any time, t , is calculated with

$$m_s[B]_s = M_{RCS}[B]_{RCS} + M_{CLA}[B]_{CLA} + m_{RWST}[B]_{RWST} + m_{IC}[B]_{IC}.$$

Sodium tetraborate is dissolved in the ice in the ice condenser. The plant technical specifications set a lower bound on boron concentration and therefore a lower bound on the sodium tetraborate concentration in the ice condenser. As sodium tetraborate is an alkaline salt (strong base and weak acid), taking a lower bound on boron concentration in the ice condenser yields lower bounds for post LOCA sump pH. For the same reason, the lower bound to the inventory in the ice condenser is taken. Ice melt vs. time m_{IC} is shown in Figure 1. The sodium concentration in the sump may be calculated from

$$m_s[Na]_s = m_{IC}[Na]_{IC}.$$

Given that the sodium in the ice condenser is in the form of sodium tetraborate,

$$[Na]_{IC} = (M_{Na}/2 M_B) [B]_{IC}$$

where M is a molecular weight.

A small concentration of lithium hydroxide in the reactor coolant (the concentration needed for chemistry control given the boron concentration assumed) is assumed. This also is assumed to be immediately transferred to the containment sump (cf. the response to Question 2).

$$m_s[\text{Li}]_s = M_{\text{RCS}}[\text{Li}]_{\text{RCS}}.$$

Hydrochloric and nitric acids may be formed in the containment under the conditions associated with a design basis LOCA, particularly in the presence of radiation. The G-values and other data pertaining to the formation of hydrochloric and nitric acids given the levels of radiation in containment are presented below (cf. the response to Question 6). From this and with some conversion, the concentration of nitrate in the sump is given as (cf. Ref. 7)

$$[\text{NO}_3^-] = 0.4526D(t)$$

where $D(t)$ is the time dependent integrated dose.

The concentration of chlorides is given as (cf. Ref. 7)

$$[\text{Cl}^-] = 35.972D(t)m_{\text{ins}}/m_s.$$

where m_{ins} is the mass of cable insulation in containment.

The upper bound to the 1 year radiation dose following a design basis LOCA is 110 Mrad. The relative radiation dose (fraction of the 1 year radiation dose) is presented in Figure 3.

Data pertaining to the inventory and transfer of water and solutes available to the containment sump are presented in Table 1 and Figures 1-3.

The spreadsheet includes the Visual Basic program PHSC as a macro. PHSC calculates solution pH using the mathematical model based on correlations and data for solutions of boric acid, sodium hydroxide, and other acids and bases from NP-5561-CCML and EPRI TR-105714 (Ref. 5, 6). The program accepts solute concentration and solution temperature as input.

2. *In the calculation of sump pH, was the sodium tetraborate in the ice condenser the only source of alkaline chemical, or were there also other alkaline chemicals such as cesium hydroxide released from the damaged core?*

Response: The calculation takes into account two sources of alkaline compounds. The bulk of the alkaline material is sodium tetraborate in the ice condenser. The calculation also accounted for a small amount of lithium hydroxide. This alkaline compound

is added to the reactor coolant during startup and maintained through the earlier phase of the fuel cycle for chemistry control. The concentration of lithium hydroxide was set to 8.5 ppm per station procedures for management of chemistry control in the reactor coolant. Test calculations showed that assuming "refueling boron" (3075 ppm) with lithium hydroxide in the reactor coolant yields lower values for post LOCA sump pH compared to setting values for end of fuel cycle (boron and lithium concentration both set to 0 ppm).

No other alkaline compounds were assumed. In particular, no fission products were taken into account.

3. *Explain the difference between the curves in Figure 1 for normalized pH at 25 C and corrected pH at T_{sump}.*

Response: Two sets of values are calculated for post LOCA containment sump pH: one at a reference temperature of 25°C and the second at the upper bound limit for post LOCA temperature of the water in the sump. The values of containment sump pH are taken at reference temperature (25°C) in evaluations of the potential for chloride induced stress corrosion cracking of stainless steel components of the ECCS and CSS. Values of containment sump pH at 25°C also are used in the evaluation of other post accident phenomena. These include post LOCA production of hydrogen in the containment atmosphere and caustic corrosion of ECCS and CSS components.

On the other hand, sump pH values at actual solution temperatures are used in assessing the effect of sump water chemistry on the transport and release of iodine isotopes following a design basis LOCA. This conforms to the method of Beahm et al. (Ref. 7) which is used in calculations of time constants for CSS washout of iodine, and iodine partitioning from leakage of Engineered Safety Features (ESF) systems in the Auxiliary Building and RWST. This method predicts that the concentration of diatomic iodine [I₂] increases with increasing concentration of hydrogen ions [H⁺] or increases with decreasing pH. The concentration of hydrogen ions increases and pH decreases with increasing solution temperature. (The concentration of hydroxyl ions [OH⁻] also increases and pOH decreases with increasing solution temperature. However, the presence of pOH is not taken into account in the calculation of the production of I₂ per the method of Beahm et al.) Thus, as currently developed, the method of Beahm et al. predicts increased formation of I₂ with increasing solution temperature. For this reason, the pH of the containment sump is calculated at the post LOCA sump water temperature as well as at 25°C.

4. *Explain what caused the dip in the Sump pH/Time curves between 7 and 46 minutes. Note: The original question was "Explain what caused the dip in the Sump pH/Time curves between 7*

and 14 minutes." In a subsequent telephone conversation, the Staff corrected the time span to "between 7 and 46 minutes."

Response: As noted above, solution pH is calculated by the EXCEL /Visual Basic program PHSC developed within Duke Energy Corporation. This program was benchmarked against a set of titration curves for a solution of sodium hydroxide and boric acid developed by Westinghouse Electric Corporation (Ref. 8). The benchmark calculations indicated that PHSC predicted pH values higher than those from the titration curves for boron concentrations of 3000 ppm or more and sodium concentrations of less than 578 ppm. The limiting difference was found to be 3%. The calculations of post LOCA sump pH indicate that following a design basis LOCA, sodium concentration in the containment sump may fall below 578 ppm between 7 and 46 minutes. (Post LOCA sump boron concentrations remain below 3000 ppm over this time span.) For this time period, the calculated values of sump pH were decreased by 3% over this time span as a conservative measure.

5. *What was the reason for limiting evaluation of pH to the first 100 minutes after a LOCA? Was it assumed that beyond 100 minutes the changes were insignificant?*

Response: Values of post LOCA containment sump pH actually were calculated for the first 3,000 minutes following the initiating event. However, the changes in the values past 100 minutes were insignificant for the following reason. The chemicals with significant effect on post LOCA sump pH are boric acid and sodium tetraborate. The entire inventory of boric acid from the RCS, cold leg accumulators, and RWST is transported to the containment sump by the end of the injection phase of the LOCA (minimum time = 13 minutes). Sodium tetraborate is transported to the containment sump with melting of the ice in the ice condenser. Ice melt is simulated to be complete at 100 minutes. Of the remaining solutes, lithium hydroxide is assumed to be transported to the containment sump at the onset of the initiating event (time 0). Nitrates and chlorides may continue to be formed under irradiation. A hand calculation of the sump pH at the end of 30 days (assumed duration of radiological consequences of the design basis LOCA) was completed. The calculation yielded a decrease of only 0.02 in the sump pH at 25°C and 0.01 in the sump pH at solution temperature compared to the values calculated at 3,000 minutes.

6. *Describe the source of hydrochloric and nitric acids in the containment sump. What are the mechanisms for generation of these acids?*

Response: Per the method of Beahm et al., the calculation of containment sump pH accounts for the formation of nitric and hydrochloric acids in the containment environment and containment sump such as may occur under irradiation following a design basis

LOCA. Nitric acid is assumed to be formed by irradiation of water and air (such as may be found in the containment and containment sump following a design basis accident). Beahm et al. do not identify a specific mechanism but instead cite two tests performed at Oak Ridge National Laboratories to evaluate the formation of nitric acid. From these experiments, the G-value for nitric acid formation is set to 0.007 molecules per 100 eV of irradiation or 7.3×10^{-6} moles/(L-Mrad). Hydrochloric acid may be formed by radiolysis or pyrolysis of chlorinated polymers. Chlorinated polymers such as Hypalon and EPR may be used in cable insulation. The amount of chlorides which may be formed with irradiation of cable insulation has been estimated to be 4.6×10^{-4} moles of HCl per lbm of insulation per Mrad, based on a G-value of 2.1 molecules per 100 eV.

References:

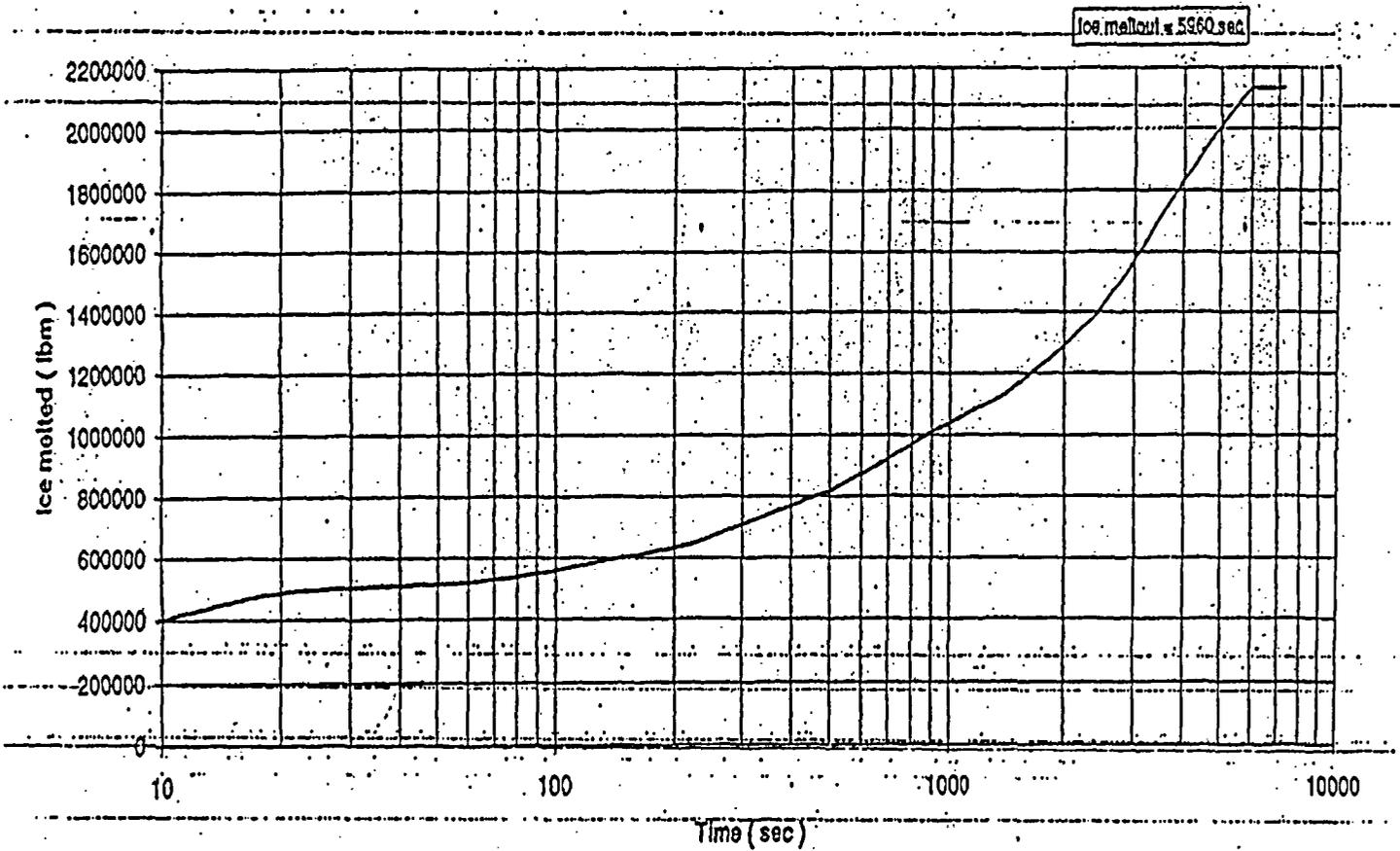
- 1) W.R. McCollum to U.S. Nuclear Regulatory Commission, "Oconee Nuclear Station Docket Numbers 50-269, 270, and 287 Supplement to License Amendment Request for Full-Scope Implementation of the Alternative Source Term Technical Specification Change (TSC) Number 2001-07," May 20, 2002.
- 2) W.R. McCollum to U.S. Nuclear Regulatory Commission, "Oconee Nuclear Station Docket Numbers 50-269, 50-270, and 50-287 License Amendment Request for Full-Scope Implementation of the Alternative Source Term and Technical Specification Technical Specifications 3.3.5, Engineered Safeguards Protective System (ESPS) Analog Instrumentation; 3.3.6, Engineered Safeguards Protective System (ESPS) Manual Initiation; 3.3.7, Engineered Safeguards Protective System (ESPS) Digital Automatic Initiation Logic Channels; 3.7.10, Penetration Room Ventilation System; 3.7.17, Spent Fuel Pool Ventilation System; 3.9.3, Containment Penetrations; 5.5.2, Containment Leakage Rate Testing Program; and 5.5.11, Ventilation Filter Testing Program Change (TSC) Number 01-07," October 16, 2001.
- 3) Catawba Nuclear Station Technical Specifications, with amendments through 200/191.
- 4) Catawba Nuclear Station Core Operating Limit Report, latest revision to date.
- 5) EPRI Report No. NP-5561-CCML, MULTEQ: Equilibrium of An Electrolytic Solution With Vapor-Liquid Partitioning and Precipitation, Volume 2: The Database (Rev 3).
- 6) EPRI Report TR-105714, PWR Primary Water Chemistry Guidelines (Rev 3).

- 7) E.C. Beahm, R.A. Lorenz, C.F. Weber, Iodine Evolution and pH Control, NUREG/CR-5950 ORNL/TM-12242, December 1992.
- 8) M.J. Bell, J.E. Bulkowski, L.J. Picone, Investigation of Chemical Additives for Reactor Containment Sprays, WCAP-7153, March 1, 1969. This report is proprietary to Westinghouse Electric Corporation. The non-proprietary version is WCAP-7803.

Table 1
Data Associated with
Containment Sump pH Following
Design Basis LOCA at Catawba Nuclear Station

Parameter	Value
Inventories	
Reactor coolant inventory (lbm)	537,793
Water volume in the RWST (gal, vortex allowance)	319,637
Lo level setpoint (gal, vortex allowance)	98,245
Cold leg accumulator water volume (cu ft)	4,320
Ice mass (lbm)	2,130,000
Cable insulation mass in containment (lbm)	16,662
Solute Concentration	
Reactor coolant boron concentration (ppm)	3,075
Cold leg accumulator boron concentration (ppm)	3,075
RWST boron concentration (ppm)	3,075
Ice condenser boron concentration (ppm)	1,800
Reactor coolant lithium concentration (ppm)	8.5
Flow Rates	
ECCS flow rate (gpm, maximum safeguards, runout)	7,895
CSS flow rate (gpm, maximum safeguards, runout)	9,600
Ice melt rate	Figure 1
Other Data	
Sump water temperature	Figure 2
Radiation dose in containment	Figure 3
Post accident HNO ₃ production rate (moles/liter-Mrad)	7.3×10^{-6}
Post accident HCl production rate (moles/lbm insulation-Mrad)	4.6×10^{-4}

CNS-2 PD Break
Ice Melted



DPC-1201.30-00-0003, CNC-1201.30-00-0029, MCC-1201.30-00-0014, Rev. 0
Catawba Ice-Melt Curve (Current Ice Weight, from Ref. 8.16)

Page 4 of 4

Figure 1 Post LOCA Ice Melt Rate vs. Time

Figure 2 Post LOCA Sump Water Temperature vs. Time

DPC-1201.30-00-0003, CNC-1201.30-00-0029, MCC-1201.30-00-0014, Rev 0
Attachment E
Page 2 of 6

Figure E-1

CNS-1 Hot Leg Break 400 lbm/basket Reduction Cases Sump Temperature

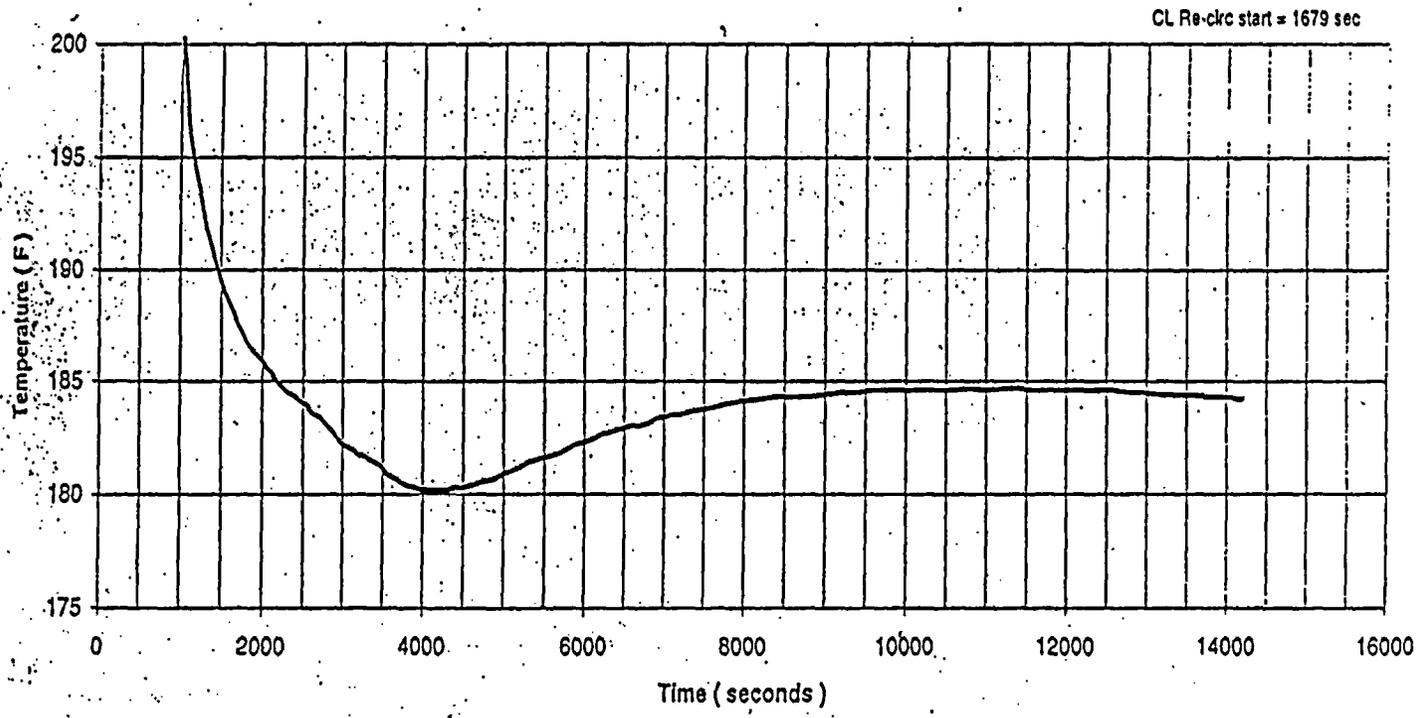


FIGURE 5.0-1

FRACTION OF ONE YEAR LOCA DOSE

Fraction Integrated Dose

USE FOR:

- 1. Auxiliary Building (Except Elev. 594 + 0)
- 2. Fuel Building (All Elev)
- 3. Reactor Building (All Elev)

Figure 3 Post Accident Integrated Gamma Dose Rate

Attachment Page 22

