UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

September 24, 2004

EA-04-115

Duke Energy Corporation
ATTN: Mr. R. A. Jones
Site Vice President
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT 05000269,270,287/2004013, OCONEE NUCLEAR STATION)

Dear Mr. Jones:

The purpose of this letter is to provide you with the Nuclear Regulatory Commission's (NRC's) final significance determination for a finding at Duke Energy Corporation's (DEC's) Oconee Nuclear Station involving fire response procedures that were not consistent with the licensing basis in regards to the criteria for manning of the Standby Shutdown Facility (SSF). In some scenarios, this could result in a delay of transfer of control to the SSF that could challenge the capability of the installed SSF makeup pump. This condition could result in the failure to maintain pressurizer level within the indicating range as required by 10 CFR 50, Appendix R.

The finding was documented in NRC Inspection Report 05000269,270,287/2004012, dated July 20, 2004, and was assessed under the significance determination process as a preliminary greater than Green issue for all three Oconee units (i.e., an issue of at least low to moderate safety significance, which may require additional NRC inspection). The cover letter to the inspection report informed DEC of the NRC's preliminary conclusion, provided DEC an opportunity to request a regulatory conference on this matter, and forwarded the details of the NRC's preliminary estimate of the change in core damage frequency (CDF) for this finding.

At your request, an open regulatory conference was conducted with members of your staff on September 13, 2004, to discuss DEC's position on this issue. The enclosures to this letter include the list of attendees at the regulatory conference, and copies of the material presented by your staff and the NRC at the regulatory conference. During the conference, DEC provided the results of its review of the safety significance of the finding and highlighted the modeling and assumption differences between its analysis of the change in CDF and that of the NRC's preliminary estimate. In addition, DEC agreed with the NRC's characterization of the finding as a violation of regulatory requirements, and stated that DEC strategy and procedures have been revised to man the SSF upon identification of a confirmed fire in the specified fire areas of concern.

A particular focus of DEC's presentation was a substantive difference in the assumed failure probability of Oconee's primary safety relief valves (PSVs). DEC stated at the conference that the NRC's PSV failure probability used in its preliminary estimate was very conservative for

Oconee's scenario. To determine a PSV failure probability that was specific to Oconee, DEC convened an Expert Elicitation Panel, commissioned with the Electric Power Research Institute. DEC explained the Expert Elicitation Panel Process in detail, and stated that its goal was to obtain a PSV "failure to reseat" probability based on test data, plant experience, and expert judgment. As described by DEC, the failure probability also considered the range of factors unique to Oconee's PSVs that may affect valve performance, such as lift type, inlet piping configuration, and fluid conditions.

Based on the efforts of the Expert Elicitation Panel, DEC concluded that the important factors in determining PSV failure rate were inlet piping configuration, fluid conditions, and the number of cycles. Regarding the factor of inlet piping configuration, DEC concluded that because the Oconee configuration is a short inlet pipe with no loop seal, its physical configuration is the most reliable relative to other piping configurations. Secondly, DEC concluded that PSV reliability is highest when relieving steam. Because the relieving fluid conditions at Oconee, for these scenarios, are expected to be steam, DEC stated that this factor would result in a higher PSV reliability relative to other fluid conditions such as water and/or subcooled liquid. Finally, DEC concluded that the PSV failure probability for cycles two through five would be substantially less than the failure probability on the initial cycle, for reasons as discussed at the conference.

Based on the above, DEC concluded that the failure probability of Oconee's PSVs was approximately one order of magnitude less than that assumed by the NRC in its preliminary estimate. As a result, DEC concluded that the finding should be characterized as Green for all three Oconee units.

After considering the information developed during the inspection and the information DEC provided at the conference, the NRC has concluded that the final inspection finding is appropriately characterized as White for all three Oconee units, in the mitigating systems cornerstone. In summary, the NRC concluded that the factors discussed at the conference are not well known with respect to their influence on PSV failure probability. The analytical techniques and risk analysis of DEC's proposal are novel and unverified with respect to the PSV failure probability following the initial lift. Additionally, DEC did not provide specific testing data to support the conclusion presented at the conference. Absent any additional specific operational, empirical, or testing data, the NRC concluded that the information provided by DEC at the conference was insufficient to warrant a change in the NRC's preliminary estimate.

You have 10 business days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC also determined that a violation occurred involving the requirements of 10 CFR 50, Appendix R, Section III.G.3, in that procedures for a fire requiring SSF manning and activation would not assure that the reactor coolant makeup function would be capable of maintaining reactor coolant level within the indicated range of the pressurizer. Accordingly, a Notice of Violation is included as an enclosure to this letter. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken to correct the violation and prevent recurrence, and the date when full compliance was achieved is adequately addressed on the docket in the information provided by DEC at the conference (Enclosure 3). Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

Based on NRC Inspection Manual Chapter 0305 guidance, the performance consideration start date for this issue is the third quarter of 2004 (i.e., when the preliminary significance determination was made known via Inspection Report 05000269,270,287/2004012, dated July 20, 2004). Consequently, as a result of this White finding, plant performance has been determined to be in the Degraded Cornerstone Column for Units 1, 2, and 3, because of a previously identified White finding in the Mitigating Systems Cornerstone (EA-03-145). We will use the NRC Action Matrix to determine the most appropriate NRC response for this finding and will notify you of that determination by separate correspondence.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures, and your response (should you choose to provide one), will be available electronically for public inspection in the NRC Public Document Room (PDR) or from the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 05000269,270,287/2004013, and the above violation is identified as VIO 05000269, 270,287/2004013-01: Failure to Meet Licensing Basis for Staffing the SSF in the Event of a Confirmed Plant Fire. Accordingly, the associated apparent violation, AV 05000269,270, 287/2004012-01, is closed.

Should you have any questions regarding this letter, please contact Charles Ogle, Chief, Division of Reactor Safety, Engineering Branch 1, at 404-562-4605.

Sincerely,

William Travers

William D. Travers
Regional Administrator

Docket Nos.: 50-269, 50-270, 50-287
License Nos.: DPR-38, DPR-47, DPR-55

Enclosures:
1. Notice of Violation
2. List of Attendees
3. Material presented by DEC
4. Material presented by NRC

cc w/encls: (see page 4)

cc w/ encls:
B. G. Davenport
Compliance Manager (ONS)
Duke Energy Corporation
Electronic Mail Distribution

Lisa Vaughn
Legal Department (PB05E)
Duke Energy Corporation
422 South Church Street
P. O. Box 1244
Charlotte, NC 28201-1244

Anne Cottingham
Winston and Strawn
Electronic Mail Distribution

Beverly Hall, Acting Director
Division of Radiation Protection
N. C. Department of Environmental
Health & Natural Resources
Electronic Mail Distribution

Henry J. Porter, Director
Div. of Radioactive Waste Mgmt.
S. C. Department of Health and
Environmental Control
Electronic Mail Distribution

R. Mike Gandy
Division of Radioactive Waste Mgmt.
S. C. Department of Health and
Environmental Control
Electronic Mail Distribution

County Supervisor of
Oconee County
415 S. Pine Street
Walhalla, SC 29691-2145

Lyle Graber, LIS
NUS Corporation
Electronic Mail Distribution

R. L. Gill, Jr., Manager
Nuclear Regulatory Licensing
Duke Energy Corporation
526 S. Church Street
Charlotte, NC 28201-0006

Peggy Force
Assistant Attorney General
N. C. Department of Justice
Electronic Mail Distribution

Distribution w/encls: (See page 5)

Distribution w/encls:
L. Reyes, EDO
W. Borchardt, NRR
L. Chandler, OGC
E. Julian, SECY
B. Keeling, OCA
Enforcement Coordinators
RI, RIII, RIV
E. Hayden, OPA
G. Caputo, OI
H. Bell, OIG
C. Carpenter, NRR
M. Johnson, NRR
R. Franovich, NRR
F. Congel, OE
W. Travers, RII
L. Plisco, RII
V. McCree, RII
L. Wert, RII
C. Casto, RII
H. Christensen, RII
W. Rogers, RII
R. Haag, RII
C. Ogle, RII
K. O'Donohue, RII
S. Sparks, RII
M. Shannon, RII
C. Evans, RII
R. Carroll, RII
R. Hannah, RII
K. Clark, RII
PUBLIC
OEMAIL
OEWEB

via D. Nelson
telephone

OFFICE	RII:DRS	RII:EICS	RII:DRP	OE	NRR	
SIGNATURE	WRogers	CEvans	CCasto for			
NAME	WROGERS	CEVANS	CCASTO	FCONGEL		
DATE	9/17/04	9/16/04	9/16/04	9/21/4		

OFFICIAL RECORD COPY DOCUMENT NAME: M:\ENFORCE\04Cases\115Oco\Oconee Final SSF.wpd

NOTICE OF VIOLATION

Duke Energy Corporatoin
Oconee Nuclear Station
Units 1, 2 and 3

Docket No.: 50-269, 50-270, 50-287
License No.:DPR-38, DPR-47, DPR-55
EA-04-115

During an NRC inspection completed on February 18, 2004, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," (Enforcement Policy), the violation is listed below:

> Oconee Unit 1 Operating License DPR-38, Oconee Unit 2 Operating License DPR-47, and Oconee Unit 3 Operating License DPR-55 Condition D provide, in part, that the licensee implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report (UFSAR) and as approved in the Safety Evaluation Report (SER) dated April 28, 1983 and subsequent supplements.
>
> The licensee's UFSAR commits to 10 CFR 50, Appendix R, Sections III.G and III.L. Section III.G.3 states that alternative shutdown capability should be provided where the protection of systems whose function is required for hot shutdown, does not satisfy the requirements of III.G.2. Section III.L of Appendix R provides requirements to be met by alternative shutdown methods. Section III.L.2.b states, in part, that "The reactor coolant makeup function shall be capable of maintaining the reactor coolant level. . . within the level indication in the pressurizer in PWRs." Section III.L.3 specifies that "procedures shall be in effect to implement this capability."
>
> Contrary to the above, on February 8, 2004, the licensee's procedures for a fire requiring SSF manning and activation would not assure that the reactor coolant makeup function would be capable of maintaining reactor coolant level within the indicated range of the pressurizer. Specifically, delaying the manning of the SSF until after the occurrence of a loss of function of the high pressure injection and component cooling or feedwater rather than manning the SSF immediately upon confirmation of a fire in the areas of concern may not preclude an extended loss of reactor coolant system inventory.

This violation is associated with a White Significance Determination Process finding for Units 1, 2 and 3.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken to correct the violation and prevent recurrence and the date when full compliance was achieved is already adequately addressed on the docket in the information provided by Duke Energy Corporation at the conference (Enclosure 3) and in NRC Inspection Report 05000269,270,287/2004012. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation - EA-04-115," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region RII, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 24th day of September 2004

LIST OF REGULATORY CONFERENCE ATTENDEES

NUCLEAR REGULATORY COMMISSION:
C. Casto, Director, Division of Reactor Safety (DRS), Region II
L. Wert, Deputy Director, Division of Reactor Projects (DRP), Region II
C. Evans, Director, Enforcement and Investigations Coordination Staff, Region II
S. Sparks, Senior Enforcement Specialist, Region II
R. Haag, Chief, Branch 1, DRP, Region II
W. Rogers, Senior Reactor Analyst, Division of Reactor Safety, Region II
C. Ogle, Chief, Engineering Branch 1, DRS, Region II
K. O'Donohue, Fire Protection Team Leader, Region II
D. Nelson, Senior Enforcement Specialist, Office of Enforcement
F. Cherny, Office of Nuclear Regulatory Research
D. Terao, Office of Nuclear Reactor Regulation (NRR)
G. Hammer, NRR
J. Lazevnick, NRR
M. Franovich, NRR
M. Tschiltz, NRR
L. Olshan, NRR
S. Long, NRR
M. Ross-Lee, NRR
S. Wong, NRR
G. Imbro, NRR

DUKE ENERGY CORPORATION:
L. Nicholson, Safety Assurance Manager
D. Baxter, Engineering Manager
D. Brewer, Nuclear Engineering Supervisor
D. Coyle, Nuclear Support Section Manager
S. Hart, Civil Structural Engineer
N. Clarkson, Senior Engineer Regulatory Compliance
K. Canavan, Electric Power Research Institute Project Engineer

PUBLIC:
Paul Gunter, Nuclear Information & Resource Service

Duke Energy℠

Safety Significance of Delayed SSF Activation During Fire

September 13, 2004

Duke Energy℠

Outline

- Characterization of Performance Deficiency – Dave Baxter
- Sequence of Events – Duncan Brewer
- Safety Relief Valve Failure Probability – Duncan Brewer
- EPRI Expert Elicitation Process – Ken Canavan
- Impact of Revised SRV Model – Duncan Brewer
- Conclusion – Dave Baxter

Duke Energy℠

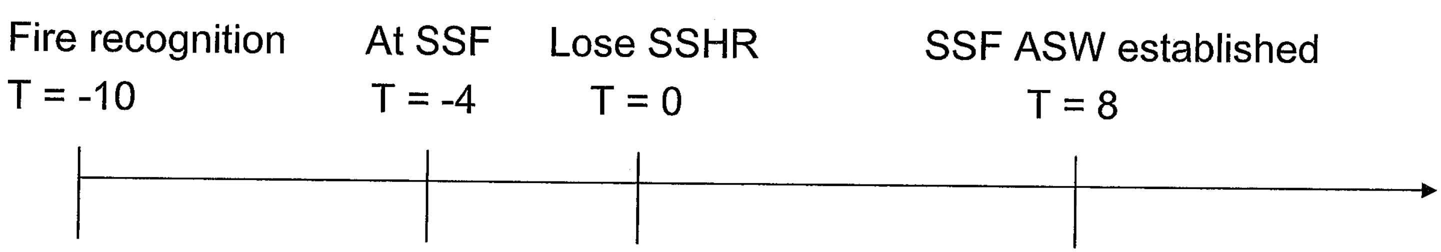
Performance Deficiency

- SSF should be manned upon identification of a fire
  - Definition of “fire” never clearly established
- Duke’s historical interpretation was to man upon loss of function
- Strategy and procedures revised to now man upon identification of a confirmed fire
- Duke does not contest the performance deficiency

Duke Energy

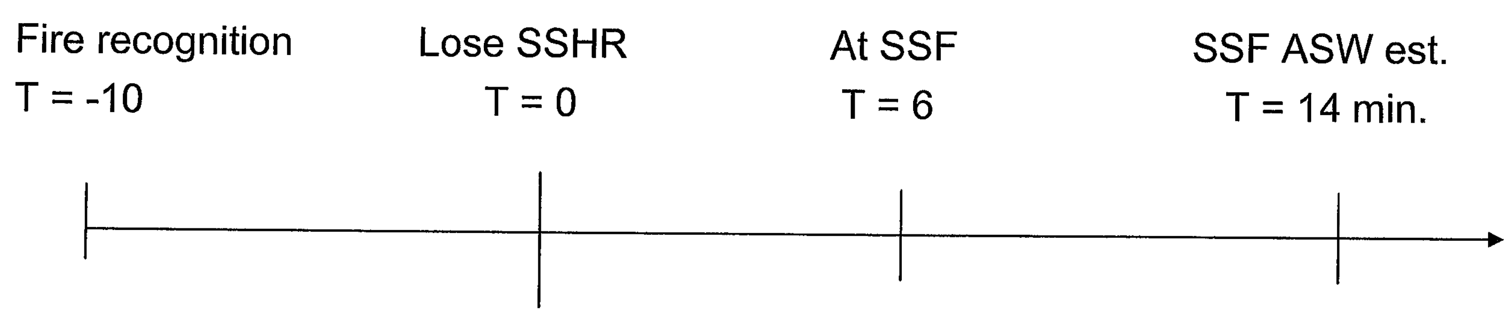
SSF Delay Time Line

- **Base Case**
  - Recognition of confirmed active fire
  - Six minutes to man the SSF from recognition of fire
  - Eight minutes to establish SG cooling after normal sources are lost



SSF Delay Time Line

- **Non Conforming Case**
  - Recognition of confirmed active fire
  - Ten minutes to lose SG cooling
  - Six minutes to man SSF from loss of function
  - Fourteen minutes to establish SG cooling after normal sources are lost



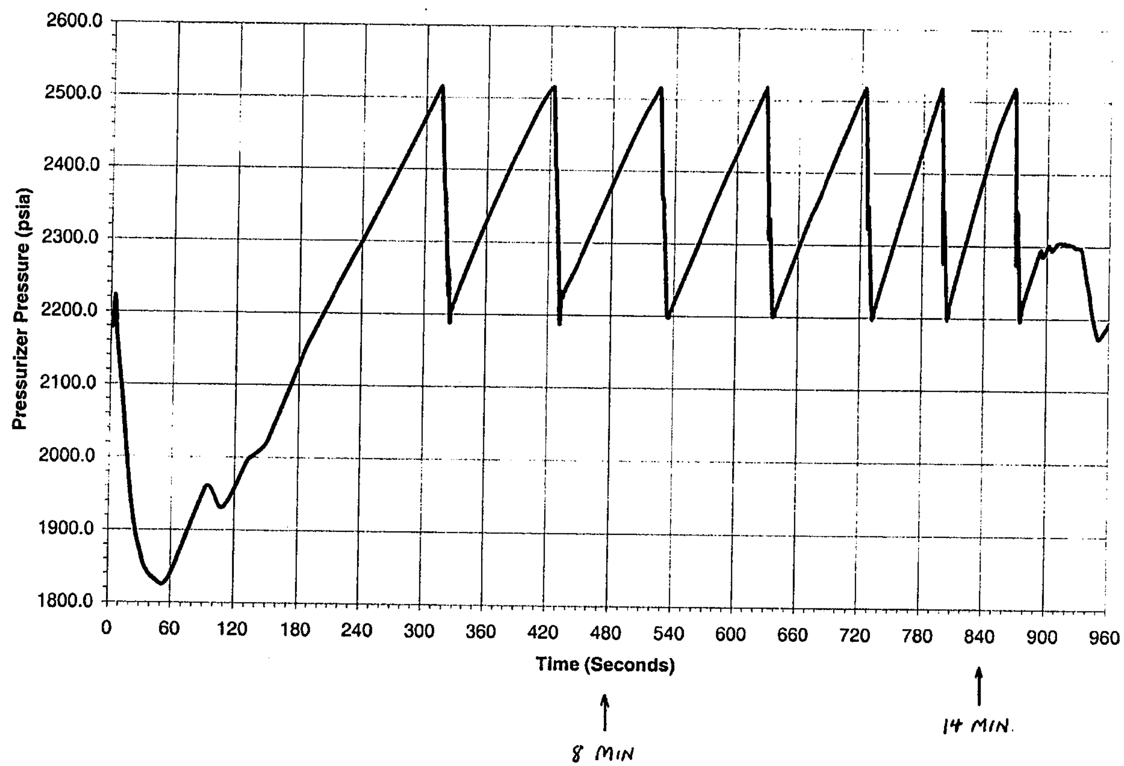
Duke Energy SM

SSF Delay T/H Issues

- Relief Valve Modeling
  - ONS Pressurizer has two safety relief valves
  - Standard licensing analyses model one valve with 2x area
  - PRA analysis assumed the lift setpoints are not identical
    - Historical as-found data confirms this
  - Only one valve cycles in this scenario
    - Pressurization rate is not very steep

T/H Results

ONS SSF Scenario
14 Minute Actuation of SSF ASW



Duke Energy℠

Primary Safety Valve Failure Rate Background

- Licensing Assumption
  - Primary Safety Valves are not subject to single failure
  - Valves open and close as designed
- PRA Assumption
  - Primary Safety Valves fail at a rate based on historical data
  - Very limited operating experience/data
- EPRI conducted testing to resolve Post-TMI relief valve issues

Duke Energy

Primary Safety Valve Failure Rate Background

- EPRI tests are single lift tests
- No industry experience with multiple lifts
- Duke and NRC have used different methods for scenarios of multiple lifts

Duke Energy℠

Primary Safety Valve Failure Rate Expert Elicitation Process

- Fall 2003 - Duke commissioned EPRI to convene an Expert Elicitation Panel
- Process is controlled by NUREG-1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program"

Duke Energy SM

Primary Safety Valve Failure Rate Expert Elicitation Process

- Ken Canavan Discussion of EPRI Report

  Probability of Safety Valve Failure to Reseat Following Steam and Liquid Relief – Quantitative Expert Elicitation

Duke Energy℠

Primary Safety Valve Failure Rate Expert Elicitation Results

- Experts concluded the important factors in determining failure rate are:
  - Inlet piping configuration
  - Fluid condition
  - Number of cycles

Duke Energy[SM]

Primary Safety Valve Failure Rate Expert Elicitation Results

- Inlet piping configuration
  - Long vs. short inlet piping
  - Existence of a loop seal
  - Oconee configuration is a short inlet pipe with no loop seal
- This is the most reliable configuration

Duke Energy[SM]

Primary Safety Valve Failure Rate Expert Elicitation Results

- Fluid conditions
  - Steam vs. water
  - Degree of subcooling
  - For this SDP issue, the relieving fluid is steam
- PSV reliability is highest when relieving steam

Duke Energy SM

Primary Safety Valve Failure Rate Expert Elicitation Results

- Number of cycles
  - Most failure modes associated with installation/manufacturing
  - Evident on first cycle
  - Subsequent cycles will have a substantially lower failure rate
  - Wear out is not a concern for this sequence
  - For this SDP evaluation, the delay in manning the SSF results in five additional RV cycles

Duke Energy

Primary Safety Valve Failure Rate Expert Elicitation Results

Lift No.	Piping Configuration	Steam Relief			Saturated Liquid (<100 F sub-cooling)			Sub-Cooled Liquid (100 - 200 F sub-cooling)			Sub-Cooled Liquid (>200 F sub-cooling)		
		Mean	5th	95th	Mean	5th	95th	Mean	5th	95th	Mean	5th	95th
First Relief	Short Inlet	2.9E-3	7.8E-4	8.1E-3	1.0E-2	2.6E-3	3.7E-2	1.8E-2	7.0E-3	7.2E-2	2.0E-1	7.3E-2	4.0E-1
	Long Inlet	3.5E-3	1.4E-3	1.1E-2	1.3E-2	2.9E-3	6.2E-2	2.7E-2	9.4E-3	1.3E-1	3.4E-1	1.2E-1	5.3E-1
	Loop Seal (hot >350 F)	2.2E-2	1.1E-2	5.1E-2	2.5E-2	7.4E-3	1.3E-1	4.8E-2	1.5E-2	1.5E-1	2.5E-1	9.4E-2	4.9E-1
	Loop Seal (cold ~100 F)	6.3E-2	2.7E-2	1.9E-1.	5.2E-2	1.3E-2	2.6E-1	7.9E-2	2.6E-2	2.1E-1	4.0E-1	2.0E-1	6.7E-1
Subsequent Reliefs	Short Inlet	5.8E-4	2.1E-4	2.5E-3	2.7E-3	1.1E-3	9.1E-3	6.3E-3	3.0E-3	1.7E-2	1.1E-1	3.9E-2	3.4E-1
	Long Inlet	1.6E-3	2.1E-4	6.9E-3	5.0E-3	1.6E-3	1.9E-2	1.2E-2	4.1E-3	3.2E-2	1.6E-1	5.4E-2	3.9E-1

Duke Energy℠

Primary Safety Valve Failure Rate Expert Elicitation Results

- Comparison of failure values
  - NRC SDP evaluation assumed probability of a safety valve sticking open is 3E-03/challenge
  - 6 cycles
  - NRC SDP failure probability = <u>1.8E-02</u>
  - EPRI study, for subsequent relief w/ steam relief, short inlet pipe, no loop seal is 5.8E-04
  - 5 cycles
  - Duke/EPRI failure probability = <u>2.9E-03</u>

Duke Energy

Results

- NRC SDP Found Greater than Green
  Delta CDF = 3E-6/yr
- NRC SRV Failure Probability (1.8E-2) is Very Conservative for ONS Scenario
- Using EPRI Method for Failure Probability and Duke Specific T/H Analysis, Failure Probability is 2.9E-3
- SDP Should be Green

Safety Valve Expert Elicitation Process

Ken Canavan

Electric Power Research Institute

September 2004

EPRI

Overview of the Expert Elicitation Process

- Expert judgment is information, provided by a technical expert in his or her subject matter area of expertise, based on opinion, or on a belief based on reasoning.
- Expert elicitation is a highly structured and well-documented process whereby expert judgments, usually of multiple experts, are obtained.
- Expert elicitation is usually used where the information cannot be answered, directly or completely, by other means.
- Expert elicitation can range from relatively informal process to an extremely formal process.
- The basis for the formality of the process is:
  - Degree the results impact the risk assessment
  - Difficulty, complexity and uncertainty of the issue
  - Controversial nature of the issue
  - Resources available
  - Public perception

EPRI

Stages of the Expert Elicitation Process

- Stage 1 – Provide the problem statement. (In the safety valve expert elicitation, stage 1 is performed via email and reviewed in the presentations.)
- Stage 2 – Bring experts together to discuss the approach to soliciting input as well as the technical issues associated with the problem statement.
- Stage 3 – Provide the experts with the results of their collective input and obtain "buy – in". In the safety valve expert elicitation, this was accomplished in a separate meeting and via review of the final report.

Expert Elicitation Process References

- The solicitation of expert opinion is based on the process as documented in the following:
  - "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts" (NUREG/CR-6372)
  - "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program" (NUREG-1563)
- The goal of the process, as applied in the Safety Valve Expert Elicitation is to obtain a safety valve failure to reseat probability based on test data, plant experience, and expert judgment. This failure probability considers the range of factors that may affect valve performance such as lift type, inlet piping configuration, and fluid conditions.

Five Functional Requirements

1. Identification of the Expert Elicitation Process
    - Define the issue (Problem Statement)
    - Determining the degree of importance (i.e., I, II, or III) and degree of complexity of the issue (A, B, C or D)
    - Deciding whether to use a Technical Integrator (TI) or Technical Facilitator / Integrator (TFI)

2. Identification and Selection of Experts
    – Experts were chosen based on experience in safety valve testing and/or maintenance and one or more of the following additional areas:
    - Safety valve tests or interpreting/ characterizing test results
    - Safety valve maintenance or development or implementation of maintenance programs for safety valves
    - Statistics / probability theory / Probabilistic Risk Assessment

Five Functional Requirements (continued)

3. Determination of the Need for Outside Expert Judgment
   – The decision was made to seek outside (i.e., expert elicitation process) expert judgment as opposed to using PRA team members due to the complexity and specificity of the estimation of the safety valve failure to reseat probability.
   – The nature of the analysis requires that technical community be involved in the development of the analysis.
4. Utilize the TI or TFI Process
   – The TFI process is applied to only Level D analysis. A Level B analysis has been chosen, therefore the Technical Integrator (TI) process is to be used. The TI process includes the following significant elements:
     - Identify available information and analysis and information retrieval methods;
     - Accumulate information relevant to the issue;
     - Perform the analysis and the data diagnostics;
     - Develop the community distribution

EPRI

Five Functional Requirements (continued)

5. Responsibility For The Expert Judgment
   - A basic principle of the expert elicitation process is a requirement that expert judgments, opinions, and/or interpretations, both as expressed by the individual experts and as integrated together have a clear defined owner.
   - In the case of the safety valve failure probability determination:
     - the owner of the process and results is the technical integrator
     - Individual expert own their individual judgments and interpretations

Expert Elicitation Degrees and Levels

Issue Degree	Decision Factors	Study Level
Degree I Non controversial; and/or insignificant to the result	Regulatory concern	Level A TI evaluates/weights models based on literature review and experience; estimates community distribution
Degree II Significant uncertainty and diversity; controversial; and complex	Resources available	Level B TI interacts with proponents and resource experts to identify issues and interpretations; estimates the community distribution
Degree III Highly contentious; significant to result and highly complex	Public perception	Level C TI brings together proponents and resource experts for debate and interaction; TI focus debate and evaluates alternative interpretations; estimate community distribution
		Level D TFI organizes panel of experts to interpret and evaluate; focus discussions; avoids inappropriate behavior on the part of the evaluators; draws picture of evaluators' estimate of the community's composite distribution; has ultimate responsibility for project

EPRI

Safety Valve Level and Degree

- A Degree II is assigned to the expert elicitation of the failure of safety valves to reseat. The basis for this assignment is as follows:
  - Significantly uncertain
  - Large impact on final risk results
  - Reasonably complex (i.e., medium complexity)
- It should be noted that although a study level of B is assigned, the elicitation was performed with a study level of C since the experts were brought together on several occasions and alternate interpretations of the data were sought.

Expert Elicitation Input Form

Lift No.	Piping Configuration	Estimate of Low, Best, & High Value	Number or Fraction of Failures to Reseat in 1000 Hypothetical Tests			
			Steam Relief	Saturated Liquid (<100 F sub-cooling)	Sub-Cooled Liquid (100 - 200 F sub-cooling)	Sub-Cooled Liquid (>200 F sub-cooling)
First Relief	Short Inlet Pipe	Low				
		"Best"				
		High				
	Long Inlet Pipe	Low				
		"Best"				
		High				
	Loop Seal (hot 350 degrees)	Low				
		"Best"				
		High				
	Loop Seal (cold - 100 degrees)	Low				
		"Best"				
		High				
Subsequent Reliefs	Short Inlet Pipe	Low				
		"Best"				
		High				
	Long Inlet Pipe	Low				
		"Best"				
		High				

- In summary, the expert elicitation process represents a significant improvement over other methods used to estimate the failure probability of safety valves to reseat.

EPRI

OPEN REGULATORY CONFERENCE

OCONEE NUCLEAR STATION

SEPTEMBER 13, 2004

NRC REGION II OFFICE, ATLANTA, GEORGIA

I. OPENING REMARKS, INTRODUCTIONS AND MEETING INTENT
   Dr. W. Travers, Regional Administrator

II. NRC REGULATORY CONFERENCE POLICY
   C. Casto, Director, Division of Reactor Safety

III. STATEMENT OF THE ISSUE WITH RISK PERSPECTIVES
   C. Casto, Director, Division of Reactor Safety

IV. SUMMARY OF APPARENT VIOLATIONS
   C. Casto, Director, Division of Reactor Safety

V. LICENSEE RISK PERSPECTIVE PRESENTATION

VI. LICENSEE RESPONSE TO APPARENT VIOLATIONS

VII. BREAK / NRC CAUCUS
   Dr. W. Travers, Regional Administrator

VIII. CLOSING REMARKS
   Dr. W. Travers, Regional Adminstrator

Enclosure 4

OPEN REGULATORY CONFERENCE

OCONEE NUCLEAR STATION

SEPTEMBER 13, 2004

NRC REGION II OFFICE, ATLANTA, GEORGIA

I. OPENING REMARKS, INTRODUCTIONS AND MEETING INTENT
   Dr. W. Travers, Regional Administrator

II. NRC REGULATORY CONFERENCE POLICY
   C. Casto, Director, Division of Reactor Safety

III. STATEMENT OF THE ISSUE WITH RISK PERSPECTIVES
   C. Casto, Director, Division of Reactor Safety

IV. SUMMARY OF APPARENT VIOLATIONS
   C. Casto, Director, Division of Reactor Safety

V. LICENSEE RISK PERSPECTIVE PRESENTATION

VI. LICENSEE RESPONSE TO APPARENT VIOLATIONS

VII. BREAK / NRC CAUCUS
   Dr. W. Travers, Regional Administrator

VIII. CLOSING REMARKS
   Dr. W. Travers, Regional Adminstrator

Enclosure 4