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Michael S. Tuckman
Executive Vice President
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November 12, 2002

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
McGuire Nuclear Station, Units 1 and 2
Docket Numbers 50-369 and 50-370
Proposed Technical Specifications (TS) Amendments
Technical Specification 5.5.2 (Containment Leakage Rate Testing Program)
One-Time Extension of Integrated Leak Rate Testing (ILRT) Interval

References:
1. Letter from Duke Energy Corporation to NRC, same subject, dated May 29, 2002
2. Letter from Duke Energy Corporation to NRC, same subject, dated September 25, 2002

In the reference letter, Duke Energy Corporation submitted a request for amendments to the Catawba and McGuire Nuclear Station Facility Operating Licenses and TS. These amendments will allow, on a one-time basis, extension of the interval governing the conduct of ILRT from ten to fifteen years.

On October 30, 2002, a conference call was held among various representatives of Duke Energy Corporation and the NRC to discuss the subject request. The purpose of this letter is to partially respond to requests for additional information raised by the NRC during the conference call. Attachment 1 to this letter provides the partial response to these requests. Additional Probabilistic Risk Assessment calculations will be necessary to provide the remainder of the response. The remainder of the response is expected to be provided by early December 2002.

A017

The original conclusions of the No Significant Hazards Consideration Analysis and the Environmental Analysis as delineated in Reference 1 are unchanged as a result of this amendment request supplement.

Pursuant to 10 CFR 50.91, copies of this letter are being sent to the appropriate state officials.

There are no regulatory commitments contained in this letter or its attachment.

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

Very truly yours,

M.S. Tuckman

M.S. Tuckman

LJR/s

Attachment

M.S. Tuckman affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

M. S. Tuckman

M.S. Tuckman, Executive Vice President

Subscribed and sworn to me: Nov 12, 2002
Date

Mary P. Nehus
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My commission expires: JAN 22, 2006
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SEAL

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SREC
Catawba Document Control File 801.01
McGuire Document Control File
Catawba RGC Date File
ELL-EC050

ATTACHMENT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (PARTIAL)

NRC Request:

Provide additional information concerning the fraction of CDF used in the development of the Person-Rem, LERF, and CCFP estimates.

Duke's Response:

The following method was used to estimate the Person-Rem risk, LERF, and CCFP frequency data:

1. Estimate the probability of Accident Classes 3a (small leak) and 3b (large, LERF leak).

2. Multiply the 3a and 3b probabilities by those Containment End-States that are potentially impacted by accident class 3a and 3b. For example, only "non-LERF" end-states are used in the LERF assessment since LERF end-states results in LERF regardless of the class 3b probability.

3. Subtract the resulting 3a and 3b frequencies from the impacted Containment End-States frequencies in order to maintain CDF.

The Containment End-State data and its associated EPRI Classification for Catawba and McGuire are provided in Tables 1 and 2.

Table 1
Catawba PRA Revision 2 Risk Results Summary

Containment End-State	Frequency (yr^{-1})	Person-Rem Risk (yr^{-1})	Person-Rem[a]	EPRI Classification
Intact Containment	2.23E-05	3.84E-02	1.72E+03	1
Basemat Melt-though	2.74E-06	1.45E-02	5.30E+03	7
Late Overpressurization Benign Failure	1.81E-06	1.32E-01	7.29E+04	7
Late Overpressurization Catastrophic Failure	1.63E-05	1.40E+01	8.62E+05	7
Early Overpressurization	3.03E-06	4.15E+00	1.37E+06	7
Large Isolation Failure	0.00E+00	0.00E+00	0.00E+00	2, 6
Small Isolation Failure[b]	1.31E-07	1.23E-02	9.41E+04	2, 6
ISLOCA	2.50E-07	2.60E+00	1.04E+07	8
Containment Bypass (SGTR)	5.16E-08	2.85E-01	5.52E+06	8
Total	4.65E-05	2.13E+01		

a. This value is the frequency-weighted person-rem. It is determined by summing the person-rem risk (rem/yr) for all release categories with the containment end-state. This sum is divided by the sum of the release category frequencies to obtain the population dose for the containment end-state.

b. Includes isolation failures due to latent human error - failure to restore the isolation following maintenance (EPRI Class 6).

Table 2
McGuire PRA Revision 2 Risk Results Summary

Containment End-State	Frequency (yr^{-1})	Person-Rem Risk (yr^{-1})	Person-Rem[a]	EPRI Classification
Intact Containment	1.72E-05	3.38E-02	1.97E+03	1
Basemat Melt-though	1.90E-06	4.13E-03	2.18E+03	7
Late Overpressurization Benign Failure	8.00E-07	4.10E-03	5.12E+03	7
Late Overpressurization Catastrophic Failure	7.20E-06	2.14E+00	2.98E+05	7
Early Overpressurization	7.43E-07	1.15E+00	1.55E+06	7
Large Isolation Failure	5.20E-08	1.13E-02	2.18E+05	2, 6
Small Isolation Failure	1.30E-09	1.16E-04	8.92E+04	2, 6
ISLOCA	2.22E-07	2.62E+00	1.18E+07	8
Containment Bypass (SGTR)	2.41E-08	1.12E-01	4.64E+06	8
Total	2.81E-05	6.08E+00		

a. This value is the frequency-weighted person-rem. It is determined by summing the person-rem risk (rem/yr) for all release categories with the containment end-state. This sum is divided by the sum of the release category frequencies to obtain the population dose for the containment end-state.

b. Includes isolation failures due to latent human error - failure to restore the isolation following maintenance (EPRI Class 6).

Catawba Person-Rem Risk Estimate:

The leakage associated with Class 3a is very small and is assumed to leak at 10 times the allowed leak rate (L_a). The CNS PRA intact containment failure (EPRI Class 1) person-rem estimates were developed for the CNS PRA assuming that containment leaks at the allowed leak rate. An estimate of the Class 3a person-rem is calculated by multiplying the Class 1 person-rem estimates by 10.

A comparison of the person-rem for Class 3a to the population dose for the other accident classes shows that the Class 3a dose is less than the dose for all accident classes except the intact containment failure class and the basemat melt-though class. Therefore, only the off-site consequences associated with these accident classes will be impacted by Class 3a leakage.

An estimate of Class 3b frequency can be made by multiplying the probability of Class 3b by the frequency of accident classes that are impacted Class 3b leakage. Class 3b is assumed to be similar to a small isolation failure. Therefore, the Class 3b population dose is assumed to be the same as the population dose for the Small Isolation Failure end-state (Table 1). This dose is 9.41E+04 person-rem.

Accident classes that are impacted by Class 3b are the accident classes with population doses less than 9.41E+04 person-rem. These accident classes consist of the late containment benign failures, the basemat melt-through failures, and the no containment failures. The Class 3b frequency is obtained by multiplying the Class 3b probability by the frequency for these containment end-states.

A sample calculation is provided in Table 3.

Table 3
Catawba Person-Rem Frequency Calculation for 1 Test in 15 Years Case

Containment End-State	Base Frequency (yr^{-1})	Class 3a Frequency (yr^{-1})	Class 3b Frequency (yr^{-1})	Revised Frequency (yr^{-1})
Intact Containment	2.23E-05	2.81E-06	2.81E-07	1.92E-05
Basemat Melt-though	2.74E-06	3.45E-07	3.45E-08	2.36E-06
Late Overpressurization Benign Failure	1.81E-06		2.28E-08	1.78E-06
Late Overpressurization Catastrophic Failure	1.63E-05			1.63E-05
Early Overpressurization	3.03E-06			3.03E-06
Large Isolation Failure	0.00E+00			0.00E+00
Small Isolation Failure	1.31E-07			1.31E-07
ISLOCA	2.50E-07			2.50E-07
Containment Bypass (SGTR)	5.16E-08			5.16E-08
Class 3a				3.15E-06
Class 3b				3.38E-07
Total	4.65E-05	3.15E-06	3.38E-07	4.65E-05

Similar calculations were performed for the other ILRT cases.

The Person-Rem risk is obtained by multiplying the Containment End-State Revised Frequency by the Containment End-State dose. A sample calculation for the 1 in 15 years case is provided in Table 4.

Table 4
Catawba Person-Rem Risk for 1 Test in 15 Years Case

Containment End-State	EPRI Class	Revised Frequency (yr^{-1})[a]	Population Dose (Person-Rem)[a]	Person-Rem (yr^{-1})[a]
Intact Containment	1	1.918E-05	1.725E+03	3.308E-02
Basemat Melt-though	7	2.357E-06	5.305E+03	1.250E-02
Late Overpressurization Benign Failure	7	1.785E-06	7.286E+04	1.300E-01
Late Overpressurization Catastrophic Failure	7	1.627E-05	8.622E+05	1.403E+01
Early Overpressurization	7	3.034E-06	1.369E+06	4.154E+00
Large Isolation Failure	2, 6	0.000E+00	0.000E+00	0.000E+00
Small Isolation Failure	2, 6	1.305E-07	9.414E+04	1.229E-02
ISLOCA	8	2.500E-07	1.040E+07	2.600E+00
Containment Bypass (SGTR)	8	5.157E-08	5.520E+06	2.847E-01
Class 3a	3a	3.150E-06	1.725E+04	5.433E-02
Class 3b	3b	3.378E-07	9.414E+04	3.180E-02
Total		4.654E-05		2.134E+01

a. Additional significant figures shown to ensure change is recognized.

Similar calculations were performed for the other ILRT cases.

McGuire Person-Rem Risk Estimate:

Similar to Catawba, the leakage associated with Class 3a for McGuire is assumed to leak at 10 times the allowed leak rate (L_a). The McGuire PRA intact containment failure (EPRI Class 1) person-rem estimates were developed for the McGuire PRA assuming that containment leaks at the allowed leak rate. An estimate of the Class 3a person-rem is calculated by multiplying the Class 1 person-rem estimates by 10.

A comparison of the person-rem for Class 3a to the population dose for the other accident classes shows that the Class 3a dose is less than the dose for all accident classes except the intact containment failure class, the basemat melt-though class, and the late overpressurization benign failure. Therefore, only the off-site consequences associated with these accident classes will be impacted by Class 3a leakage.

An estimate of Class 3b frequency can be made by multiplying the probability of Class 3b by the frequency of accident classes that are impacted Class 3b leakage. Class 3b is assumed to be similar to a small isolation failure.

Therefore, the Class 3b population dose is assumed to be the same as the population dose for the Small Isolation Failure end-state (Table 2). This dose is 8.92E+04 person-rem.

Accident classes that are impacted by Class 3b are the accident classes with population doses less than 8.92E+04 person-rem. These accident classes consist of the late containment benign failures, the basemat melt-through failures, and the no containment failures. The Class 3b frequency is obtained by multiplying the Class 3b probability by the frequency for these containment end-states.

A sample calculation is provided in Table 5.

Table 5
McGuire Person-Rem Frequency Calculation for 1 Test in 15 Years Case

Containment End-State	Base Frequency (yr^{-1})	Class 3a Frequency (yr^{-1})	Class 3b Frequency (yr^{-1})	Revised Frequency (yr^{-1})
Intact Containment	1.72E-05	2.17E-06	2.17E-07	1.48E-05
Basemat Melt-though	1.90E-06	2.39E-07	2.39E-08	1.64E-06
Late Overpressurization Benign Failure	8.00E-07	1.01E-07	1.01E-08	6.89E-07
Late Overpressurization Catastrophic Failure	7.20E-06			7.20E-06
Early Overpressurization	7.43E-07			7.43E-07
Large Isolation Failure	5.20E-08			5.20E-08
Small Isolation Failure	1.30E-09			1.30E-09
ISLOCA	2.22E-07			2.22E-07
Containment Bypass (SGTR)	2.41E-08			2.41E-08
Class 3a				2.51E-06
Class 3b				2.51E-07
Total	2.81E-05	2.51E-06	2.51E-07	2.81E-05

Similar calculations were performed for the other ILRT cases.

The Person-Rem risk is obtained by multiplying the Containment End-State Revised Frequency by the Containment End-State dose. A sample calculation for the 1 in 15 years case is provided in Table 6.

Table 6
McGuire Person-Rem Risk for 1 Test in 15 Years Case

Containment End-State	EPRI Class	Revised Frequency (yr^{-1})[a]	Population Dose (Person-Rem)[a]	Person-Rem (yr^{-1})[a]
Intact Containment	1	1.481E-05	1.968E+03	2.913E-02
Basemat Melt-though	7	1.635E-06	2.178E+03	3.561E-03
Late Overpressurization Benign Failure	7	6.893E-07	5.123E+03	3.532E-03
Late Overpressurization Catastrophic Failure	7	7.200E-06	2.979E+05	2.145E+00
Early Overpressurization	7	7.432E-07	1.549E+06	1.151E+00
Large Isolation Failure	2, 6	5.203E-08	2.176E+05	1.132E-02
Small Isolation Failure	2, 6	1.300E-09	8.916E+04	1.159E-04
ISLOCA	8	2.220E-07	1.180E+07	2.620E+00
Containment Bypass (SGTR)	8	2.415E-08	4.639E+06	1.120E-01
Class 3a	3a	2.506E-06	1.968E+04	4.931E-02
Class 3b	3b	2.506E-07	8.916E+04	2.234E-02
Total		2.813E-05		6.147E+00

a. Additional significant figures shown to ensure change is recognized.

Similar calculations were performed for the other ILRT cases.

LERF Estimate:

Class 3a is assumed to be too small to result in LERF. Class 3b is assumed to result in LERF. Since Class 3b represents LERF, an estimate of LERF can be made by multiplying the probability of Class 3b by the frequency of accident classes that are not LERF. This estimate is conservative since a portion of the accident sequences that are not currently LERF, may not meet the early requirements for LERF even with a Class 3b failure.

For Catawba, LERF is 3.47E-06/yr and the "non-LERF" CDF is 4.31E-05/yr (4.65E-05 - 3.47E-06/yr). For McGuire, LERF is 1.04E-06/yr and the "non-LERF" CDF is 2.71E-05/yr (2.81E-05 - 1.04E-06/yr).

The LERF for the 1 in 15 years case is:

Catawba
4.31E-05/yr x 1.26E-02 = 5.43E-07 (Class 3b)
3.47E-06/yr + 5.43E-07 = 4.01E-06 (Total LERF, Internal Events Only)

McGuire
2.71E-05/yr x 1.26E-02 = 3.41E-07 (Class 3b)
1.04E-06/yr + 3.41E-07 = 1.38E-06 (Total LERF, Internal Events Only)

Similar calculations were performed for the other ILRT cases.

CCFP Estimate:

The NRC uses Conditional Containment Failure Probability (CCFP) as an independent measure of containment mitigation capability. This probability is expressed by the following equation:

$$CCFP = 1 - \frac{ncf}{CDF}$$

ncf = no containment failure frequency
CDF = Core Damage Frequency

The no containment failure frequency is represented by accident Class 1 (No Containment Failure End-State) and the portion of Class 3a that is based on Class 1. Recall that the Class 3a frequency was calculated in the Person-Rem analysis by multiplying the Class 3a probability by the intact containment frequency and the basemat melt-though frequency for Catawba and the intact containment frequency, the basemat melt-though frequency, and the late benign failure frequency for McGuire. The basemat melt-through and the late benign failure portions of Class 3a should not be included in the No Containment Failure frequency since for these classes, containment has failed.

The no containment failure frequency for the CCFP equation is calculated by adding the revised No Containment Failure frequency (e.g., Tables 3 & 4) and the probability of Class 3a times the PRA Intact Containment frequency. For the 1 test in 15 years case, the No Containment Failure frequency and CCFP is:

Catawba:

ncf = Revised Intact Freq. + PRA Intact Containment Freq. x $Prob_{3a}$

ncf = 1.918E-05/yr + (2.23E-05/yr x 0.126)

ncf = 2.198E-05/yr

$$CCFP = 1 - \frac{2.198E\text{-}05/yr}{4.65E\text{-}05} = 52.77\%$$

McGuire:

ncf = Revised Intact Freq. + PRA Intact Containment Freq. x $Prob_{3a}$
ncf = 1.481E-05/yr + (1.72E-05/yr x 0.126)
ncf = 1.697E-05/yr

$$CCFP = 1 - \frac{1.697E\text{-}05/yr}{2.81E\text{-}05} = 39.66\%$$

Similar calculations were performed for the other ILRT cases.

NRC Request:

Provide statements concerning PRA quality for Catawba and McGuire Nuclear Stations.

Duke's Response:

Catawba PRA Quality

PRA Updates

Duke's Severe Accident Analysis Group (SAAG) periodically evaluates changes to the plant with respect to the assumptions and modeling in the Catawba PRA. The original Catawba PRA was initiated in July 1984 by Duke Power Company assisted by several outside contractors who performed specialized subtasks. It was a full scope Level 3 PRA with internal and external events. A peer review sponsored by the Electric Power Research Institute (EPRI) was conducted after completion of the draft report. The study was published in an internal Duke report (Ref. 1) in 1987 as Revision 0 to the PRA.

On November 23, 1988, the NRC issued Generic Letter 88-20 (Ref. 2), which requested that licensees conduct an Individual Plant Examination (IPE) in order to identify potential severe accident vulnerabilities at their plants. The Catawba response to GL 88-20 was provided by letter dated September 10, 1992 (Ref. 3). Catawba's response included an updated Catawba PRA (Revision 1) study.

The Catawba PRA Revision 1 study and the IPE process resulted in a comprehensive, systematic examination of Catawba with regard to potential severe accidents. The Catawba study was again a full-scope, Level 3 PRA with analysis of both the internal and external events. This examination identified the most likely severe accident sequences, both internally and externally induced, with quantitative perspectives on likelihood and fission product release potential. The results of the study prompted changes in equipment, plant configuration and enhancements in plant procedures to reduce vulnerability of the plant to some accident sequences of concern.

By letter dated June 7, 1994 (Ref. 4), the NRC provided a Safety Evaluation of the internal events portion of the above Catawba IPE submittal. The conclusion of the NRC letter [page 16] states:

"The staff finds the licensee's IPE submittal for internal events including internal flooding essentially complete, with the level of detail consistent with the information requested in NUREG-1335. Based on the review of the submittal and the associated supporting information, the staff finds reasonable the licensee's IPE conclusion that no fundamental weakness or severe accident vulnerabilities exist at Catawba."

In response to Generic Letter 88-20, Supplement 4, Duke completed an Individual Plant Examination of External Events (IPEEE) for severe accidents. This IPEEE was submitted to the NRC by letter dated June 21, 1994 (Ref. 5). The report contained a summary of the methods, results and conclusions of the Catawba IPEEE program. The IPEEE process and supporting Catawba PRA included a comprehensive, systematic examination of severe accident potential resulting from external initiating events. By letter dated April 12, 1999 (Ref. 6), the NRC provided an evaluation of the IPEEE submittal. The conclusion of the NRC letter [page 6] states:

"The staff finds the licensee's IPEEE submittal is complete with regard to the information requested by Supplement 4 to GL 88-20 (and associated guidance in NUREG-1407), and the IPEEE results are reasonable given the Catawba design, operation, and history. Therefore, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the Catawba IPEEE has met the intent of Supplement 4 to GL 88-20."

In 1996, Catawba initiated Revision 2 of the 1992 IPE and provided the results to the NRC in 1998 (Ref. 7). In April 2001 Duke notified the NRC (Ref. 8) that a voluntary initiative at the Catawba Nuclear Station to provide backup cooling to the 1A and 2A high head safety injection Centrifugal Charging (NV) Pumps had been completed. In conjunction with the completion of the plant modifications, the Catawba PRA Level 1 analysis was also updated and was designated as Revision 2b. The impact of this modification was to lower the base case CDF. Revision 2b was used for the analysis supporting the extension of the ILRT period.

Currently, Revision 3 of the Catawba PRA is underway. This update, which is a comprehensive revision to the PRA models and associated documentation, is expected to be completed in 2003.

PRA maintenance encompasses the identification and evaluation of new information into the PRA and typically involves minor modifications to the plant model. PRA maintenance and updates as well as guidance for developing PRA data and evaluation of plant modifications, are governed by Workplace Procedures. In January 2001, an enhanced manual configuration control process was implemented to more effectively track, evaluate, and implement PRA changes to better ensure the PRA reflects the as-built, as-operated plant. This process was further enhanced in July 2002 with the implementation of an electronic PRA change tracking tool.

Peer Review Process

Between March 18-22, 2002, Catawba participated in the Westinghouse Owners Group (WOG) PRA Certification Program. This review followed a process that was originally developed and used by the Boiling Water Reactor Owners Group (BWROG) and subsequently broadened to be an industry-applicable process through the Nuclear Energy Institute (NEI) Risk Applications Task Force. The resulting industry document, NEI-00-02 (Ref. 9), describes the overall PRA peer review process. The Certification/Peer Review process is also linked to the ASME PRA Standard (Ref. 10).

NEI has developed draft guidance for self-assessments to address the use of industry peer review results in demonstrating conformance with the ASME PRA standard. This guidance supplements, and is expected to ultimately become part of, NEI-00-02, PRA Peer Review Process Guidance. The guidance is intended to support development of NRC draft regulatory guide DG-1122 (Determining Technical Adequacy of PRA Results for Risk-Informed Activities) which will endorse the ASME standard and discuss the industry peer review process as a means of addressing the requirements of the standard.

The objective of the PRA Peer Review process is to provide a method for establishing the technical quality and adequacy of a PRA for a range of potential risk-informed plant applications for which the PRA may be used. The PRA Peer Review process employs a team of PRA and system analysts, who possess significant expertise in PRA development and PRA applications. The team uses checklists to evaluate the scope, comprehensiveness, completeness, and fidelity of the PRA being reviewed. One of the key parts of the review is an assessment of the maintenance and update process to ensure the PRA reflects the as-built plant.

The review team for the Catawba PRA Peer Review consisted of six members. Three of the members were PRA personnel from other utilities. The remaining three were industry consultants. Reviewer independence was maintained by assuring that none of the six individuals had any involvement in the development of the Catawba PRA or IPE.

A summary of some of the Catawba PRA strengths and recommended areas for improvement from the peer review are as follows:

Strengths

- Aggressive response to past PRA peer reviews
- Knowledgeable personnel
- Culture of continuous improvement
- Documentation of final results and analyses
- Good capture of plant experience into the model
- Rigorous Level 2 and 3 PRA

Recommended Areas for Improvement

- Limited comparison to other plant/utilities PRAs for results and techniques
- Better documentation of bases for success criteria and Human Reliability Analysis timing
- More focus on realism vs. conservatism in models
- More attention to eliminating old documentation and modeling assumptions/simplifications
- Consider more efficient methods to streamline recovery/post-processing process

The significance levels of the WOG Peer Review Certification process have the following definitions:

A. Extremely important and necessary to address to ensure the technical adequacy of the PRA, the quality of the PRA, or the quality of the PRA update process.

B. Important and necessary to address but may be deferred until the next PRA update.

Based on the draft PRA peer review report, the Catawba PRA received no "A" fact and observation findings but did receive 30 "B" fact and observation findings. The "B" findings have been reviewed and prioritized for incorporation into the PRA. Some of the "B" findings have already been resolved.

Results of Reviews with Respect to this License Amendment Request

Consistent with the work place procedures governing PRA analysis, this calculation has undergone independent checking by a qualified reviewer. Additionally the Catawba Plant Operations Review Committee (PORC) and Duke Nuclear Safety Review Board (NSRB) reviewed and approved the original amendment request package.

PRA Quality Assurance Methods

Approved workplace procedures address the quality assurance of the PRA. One way the quality assurance of the Catawba PRA is ensured is by maintaining a set of system notebooks on each of the PRA systems. Each system PRA analyst is responsible for updating a specific system model. This update consists of a comprehensive review of the system including drawings and plant modifications made since the last update as well as implementation of any PRA change notices that may exist on the system. The analyst's primary focal point is with the system engineer at the site. The system engineer provides information for the update as needed. The analyst will review the PRA model with the system engineer and as necessary, conduct a system walkdown with the system engineer. This interaction is documented in a memorandum.

The system notebooks contain, but are not limited to, documentation on system design, testing and maintenance practices, success criteria, assumptions, descriptions of the reliability data, as well as the results of the quantification. The system notebooks are reviewed and signed off by a second independent person and are approved by the manager of the group.

When any change to the PRA is identified, the same three-signature process of identification, review, and approval is utilized to ensure that the change is valid and that it receives the proper priority.

References:

1. "Catawba Nuclear Station Unit 1 Probabilistic Risk Assessment," Volumes 1-3, Duke Power Company, August 18, 1987.

2. Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities, USNRC, November 1988.

3. Letter Duke Power Company to Document Control Desk (USNRC), Catawba Units 1 and 2, "Individual Plant Examination (IPE) Submittal in Response to Generic Letter 88-20," September 10, 1992.

4. Letter USNRC to Duke Power Company, "Safety Evaluation of Catawba Nuclear Station, Units 1 and 2 Individual Plant Examination (IPE) Submittal," June 7, 1994.

5. Letter Duke Power Company to Document Control Desk (USNRC), Catawba Units 1 and 2, "Individual Plant Examination of External Events (IPEEE) Submittal," June 21, 1994.

6. Letter USNRC to Duke Power Company, "CATAWBA NUCLEAR STATION -- REVIEW OF INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)," April 12, 1999.

7. Letter Duke Energy Corporation to Document Control Desk (USNRC), Catawba Units 1 and 2, "Probabilistic Risk Assessment (PRA), Revision 2 Summary Report, January 1998."

8. Letter Duke Energy Corporation to Document Control Desk (USNRC), Catawba Units 1 and 2, "Centrifugal Charging Pump Modifications and Catawba PRA Update (Revision 2b)," April 18, 2001.

9. NEI-00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guideline," Nuclear Energy Institute, January 2000.

10. "Standard For Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-S-2002, January 31, 2002.

NRC Request:

Provide statements concerning PRA quality for Catawba and McGuire Nuclear Stations.

Duke's Response:

McGuire PRA Quality

PRA Updates

Duke's Severe Accident Analysis Group (SAAG) periodically evaluates changes to the plant with respect to the assumptions and modeling in the McGuire PRA. The original McGuire PRA was initiated in March 1982 by Duke Power Company staff with Technology for Energy Corporation as a contractor. Law Engineering Testing Company and Structural Mechanics Associates provided specific input to the seismic analysis. It was a full scope Level 3 PRA with internal and external events. A peer review of the draft PRA was conducted by Electric Power Research Institute's Nuclear Safety Analysis Center (NSAC) in May 1983 (Ref. 1). The final study, which incorporated the comments of the peer review, was completed in July 1984 and resulted in an internal Duke report (Ref. 2) as Revision 0 to the PRA. In January 1988, Duke Power Company initiated a complete review and update of the original study.

On November 23, 1988, the NRC issued Generic Letter 88-20 (Ref. 3), which requested that licensees conduct an Individual Plant Examination (IPE) in order to identify potential severe accident vulnerabilities at their plants. The McGuire response to GL 88-20 was provided by letter dated November 4, 1991 (Ref. 4). McGuire's response included an updated McGuire PRA (Revision 1) study which was the culmination of the review and update which began in January 1988.

The McGuire PRA Revision 1 study and the IPE process resulted in a comprehensive, systematic examination of McGuire with regard to potential severe accidents. The McGuire study was again a full-scope, Level 3 PRA with analysis of both the internal and external events. This examination identified the most likely severe accident sequences, both internally and externally induced, with quantitative perspectives on likelihood and fission product release potential. The results of the study prompted changes in equipment, plant configuration and enhancements in plant procedures to reduce vulnerability of the plant to some accident sequences of concern.

As part of the Generic Letter 88-20 IPE process, the NRC conducted an audit of the human reliability analysis of the McGuire IPE during the period July 28 - 30, 1993. By letter dated June 30, 1994 (Ref. 5), the NRC provided a Staff Evaluation of the internal events portion of the above McGuire IPE submittal which included the results of the human reliability analysis audit. The conclusion of the NRC letter [page 15] states:

"The staff finds the licensee's IPE submittal for internal events including internal flooding essentially complete, with the level of detail consistent with the information requested in NUREG-1335. Based on the review of the submittal, and audit of "tier 2" supporting information, the staff finds reasonable the licensee's IPE conclusion that no severe accident vulnerabilities exist at McGuire."

In response to Generic Letter 88-20, Supplement 4, Duke completed an Individual Plant Examination of External Events (IPEEE) for severe accidents. This IPEEE was submitted to the NRC by letter dated June 1, 1994 (Ref. 6). The report contained a summary of the methods, results and conclusions of the McGuire IPEEE program. The IPEEE process and supporting McGuire PRA included a comprehensive, systematic examination of severe accident potential resulting from external initiating events. By letter dated February 16, 1999 (Ref. 7), the NRC provided an evaluation of the IPEEE submittal. The conclusion of the NRC letter [page 6] states:

"On the basis of the overall review findings, the staff concludes that: (1) the licensee's IPEEE is complete with regard to the information requested by Supplement 4 to GL 88-20 (and associated guidance in NUREG-1407), and (2) the IPEEE results are reasonable given the MNS design, operation, and history. Therefore, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the MNS IPEEE has met the intent of Supplement 4 to GL 88-20 and the resolution of specific generic safety issues discussed in the SER."

In 1997, McGuire initiated Revision 2 of the 1991 IPE and provided the results to the NRC in 1998 (Ref. 8). Revision 2 was used for the analysis supporting the extension of the ILRT period.

The Level 1 analysis for Revision 3 of the McGuire PRA was completed in July 2002. Level 2 and 3 analyses are currently underway.

PRA maintenance encompasses the identification and evaluation of new information into the PRA and typically involves minor modifications to the plant model. PRA maintenance and updates as well as guidance for developing PRA data and evaluation of plant modifications, are governed by Workplace Procedures. In January 2001, an enhanced manual configuration control process was implemented to more effectively track, evaluate, and implement PRA changes to better ensure the PRA reflects the as-built, as-operated plant. This process was further enhanced in July 2002 with the implementation of an electronic PRA change tracking tool.

Peer Review Process

Between October 23-27, 2000, McGuire participated in the Westinghouse Owners Group (WOG) PRA Certification Program. This review followed a process that was originally developed and used by the Boiling Water Reactor Owners Group (BWROG) and subsequently broadened to be an industry-applicable process through the Nuclear Energy Institute (NEI) Risk Applications Task Force. The resulting industry document, NEI-00-02 (Ref. 9), describes the overall PRA peer review process. The Certification/Peer Review process is also linked to the ASME PRA Standard (Ref. 10).

NEI has developed draft guidance for self-assessments to address the use of industry peer review results in demonstrating conformance with the ASME PRA standard. This guidance supplements, and is expected to ultimately become part of, NEI-00-02, PRA Peer Review Process Guidance. The guidance is intended to support development of NRC draft regulatory guide DG-1122 (Determining Technical Adequacy of PRA Results for Risk-Informed Activities) which will endorse the ASME standard and discuss the industry peer review process as a means of addressing the requirements of the standard.

The objective of the PRA Peer Review process is to provide a method for establishing the technical quality and adequacy of a PRA for a range of potential risk-informed plant applications for which the PRA may be used. The PRA Peer Review process employs a team of PRA and system analysts, who possess significant expertise in PRA development and PRA applications. The team uses checklists to evaluate the scope, comprehensiveness, completeness, and fidelity of the PRA being reviewed. One of the key parts of the review is an assessment of the maintenance and update process to ensure the PRA reflects the as-built plant.

The review team for the McGuire PRA Peer Review consisted of six members. Three of the members were PRA personnel from other utilities. The remaining three were industry consultants. Reviewer independence was maintained by assuring that none of the six individuals had any involvement in the development of the McGuire PRA or IPE.

A summary of some of the McGuire PRA strengths and recommended areas for improvement from the peer review are as follows:

Strengths

- Good Summary Report write-up with insights
- Good system notebooks
- Rigorous Level 2 & 3 PRA Model
- Integrated internal and external events model
- Up-to-date plant database using Maintenance Rule
- Ongoing PRA staff interaction with plant staff, plant staff reviews
- PRA knowledge of plant good

Recommended Areas for Improvement

- Better integration of sequences and recoveries within quantification process needed
- Need to review treatment of events requiring time-phasing in the modeling
- Better approach to closing the loop on PRA update items (tracking of errors/modifications) needed
- More thorough, systematic approach to Human Reliability Analysis screening values and common cause modeling needed
- Need an approach for reconciling realistic LERF model with NRC expectations from simplistic LERF modeling
- Need to update the PRA model to be more in line with current practices and expectations for state-of-the-art PRA

The significance levels of the WOG Peer Review Certification process have the following definitions:

A. Extremely important and necessary to address to ensure the technical adequacy of the PRA, the quality of the PRA, or the quality of the PRA update process.

B. Important and necessary to address but may be deferred until the next PRA update.

Based on the draft PRA peer review report, the McGuire PRA received six "A"s and 31 "B"s. All six of the "A"s have been resolved and changes have been incorporated into Revision 3 of the McGuire PRA. The "B" findings have been reviewed and prioritized for incorporation into the PRA. Some of the "B" findings have already been incorporated into Revision 3 of the PRA. As evidence of the effectiveness of the enhancements made to the PRA, it should be noted that the Catawba Nuclear Station, which can be considered a "sister" plant to McGuire, received its PRA Certification March 18-22, 2002 and received no "A" findings.

Results of Reviews with Respect to this License Amendment Request

Consistent with the work place procedures governing PRA analysis, this calculation has undergone independent checking by a qualified reviewer. Additionally the McGuire Plant Operations Review Committee (PORC) and Duke Nuclear Safety Review Board (NSRB) reviewed and approved the original amendment request package.

PRA Quality Assurance Methods

Approved workplace procedures address the quality assurance of the PRA. One way the quality assurance of the McGuire PRA is ensured is by maintaining a set of system notebooks on each of the PRA systems. Each system PRA analyst is responsible for updating a specific system model. This update consists of a comprehensive review of the system including drawings and plant modifications made since the last update as well as implementation of any PRA change notices that may exist on the system. The analyst's primary focal point is with the system engineer at the site. The system engineer provides information for the update as needed. The analyst will review the PRA model with the system engineer and as necessary, conduct a system walkdown with the system engineer. This interaction is documented in a memorandum.

The system notebooks contain, but are not limited to, documentation on system design, testing and maintenance practices, success criteria, assumptions, descriptions of the reliability data, as well as the results of the quantification. The system notebooks are reviewed and signed off by a second independent person and are approved by the manager of the group.

When any change to the PRA is identified, the same three-signature process of identification, review, and approval is utilized to ensure that the change is valid and that it receives the proper priority.

References:

1. Nuclear Safety Analysis Center, "McGuire Unit 1 PRA Peer Review," May 27, 1983.

2. "McGuire Nuclear Station Unit 1 Probabilistic Risk Assessment," Volumes 1-2, Duke Power Company, July 1984.

3. Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities, USNRC, November 1988.

4. Letter Duke Power Company to Document Control Desk (USNRC), McGuire Nuclear Station, "Generic Letter 88-20," November 4, 1991.

5. Letter USNRC to Duke Power Company, "Staff Evaluation of the McGuire Nuclear Station, Units 1 and 2 Individual Plant Examination - Internal Events Only," June 30, 1994.

6. Letter Duke Power Company to Document Control Desk (USNRC), McGuire Nuclear Station, Units 1 and 2, "Individual Plant Examination of External Events (IPEEE) Submittal," June 1, 1994.

7. Letter USNRC to Duke Power Company, "REVIEW OF MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 - INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS SUBMITTAL," February 16, 1999.

8. Letter Duke Energy Corporation to Document Control Desk (USNRC), McGuire Nuclear Station, "1997 Update of Probabilistic Risk Assessment," March 19, 1998.

9. NEI-00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guideline," Nuclear Energy Institute, January 2000.

10. "Standard For Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-S-2002, January 31, 2002.