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ACCESSION NBR:8911060276 DOC.DATE: 89/10/24 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315
50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316

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SUBJECT: Forwards description of individual plant exam program,per Generic Ltr 88-20.

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Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
INDIVIDUAL PLANT EXAMINATION PROGRAM PLAN,
RESPONSE TO GENERIC LETTER NO. 88-20, SUPPLEMENT NO. 1

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

Attn: T. E. Murley

October 24, 1989

Dear Dr. Murley:

In November 1988, the NRC issued Generic Letter 88-20 to formalize the requirement for Individual Plant Examination (IPE) for operating licensees. Specifically, the generic letter required licensees to perform an IPE and identify potential improvements to address important contributors to risk and implement necessary improvements. Subsequently, in August 1989, the NRC announced the availability of NUREG-1335, "Individual Plant Examination: Submittal Guidance," in Supplement 1 to the generic letter. Supplement 1 also required that licensees submit their proposed programs for conducting the IPE. The attachment to this letter provides the description of the IPE program pursuant to that request for the Donald C. Cook Nuclear Plant Units 1 and 2.

This document has been prepared following Corporate procedures that incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

M. P. Alexich
Vice President

ldp

Attachment

A003
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cc: D. H. Williams, Jr.
A. A. Blind - Bridgman
R. C. Callen
G. Charnoff
A. B. Davis
NRC Resident Inspector - Bridgman
NFEM Section Chief

INDIVIDUAL PLANT EXAMINATION
DONALD C. COOK NUCLEAR PLANT
PROGRAM DESCRIPTION

Pursuant to the requirements of Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," and its Supplement 1, the following information is provided. Specifically, this description of the IPE for the Donald C. Cook Nuclear Plant is divided into three sections, 1) Project Approach, 2) Project Methodology, and 3) Project Schedule.

1.0 Project Approach

The American Electric Power Service Corporation (AEPSC) has established a series of specific objectives for the severe accident issue resolution program for the Cook Nuclear Plant. The program objectives are:

1. To identify, evaluate and resolve the severe accident issues germane to the Cook Nuclear Plant in a realistic, technically acceptable manner with emphasis on the prevention of such accidents.

2. To identify and develop input to decision making processes relative to potential enhancements to plant design and/or operation aimed at reduction of risk from severe accidents.

3. To evolve a realistic, documented, auditable Probabilistic Risk Assessment (PRA) for the Cook Nuclear Plant which can be readily used and maintained, and which will be suitable for on-going use.

4. To address the existing NRC information request in Generic Letter 88-20 and those information requests anticipated in the near future on closely related topics (e.g., external events IPE and accident management).

5. To familiarize AEPSC and Cook Nuclear Plant staffs with the basis and methodology of PRA so that, in the future, the PRA can be independently maintained and updated as necessary.

AEPSC plans to implement these objectives through the performance of a Level III PRA considering both internal and external events. The PRA will be performed with realistic assessments of the plant's behavior under severe accident conditions and will initially be based on the plant's configuration and operating procedures in effect on August 1, 1989. AEPSC and Cook Nuclear Plant personnel

will be actively involved in this process as contributors to the analysis on a large scale and as technical and operational reviewers. In addition, AEPSC plans to comply with the NRC's stated request to have USI A-45, Decay Heat Removal, and the 10 CFR 50.44(c) hydrogen control issue addressed and resolved in the PRA process. The scope of the Cook Nuclear Plant PRA Project will not address anticipated accident management issues; however, the technical elements and documentation will be structured to provide the flexibility for ready interface with a future accident management program.

The Cook Nuclear Plant PRA Project will be performed and managed by AEPSC with support from the Individual Plant Evaluation Partnership (IPEP). The IPEP is comprised of Westinghouse Electric Corporation, Fauske and Associates, and TENERA. Members of these organizations were directly responsible for the management of the Industry Degraded Core Rulemaking (IDCOR) Program and were the primary contractors used by IDCOR to develop the industry's technical positions on severe accident issues.

The PRA team will be responsible for managing and performing the technical evaluations, as well as providing the technology transfer to other parts of the AEPSC and Cook Nuclear Plant organization. IPEP's responsibility will be to guide the creation of the system, event tree, data, and other notebooks, the performance of the initial quantifications, and the creation of reports. IPEP will provide guidance on key methodologies and their effectiveness in previous PRAs. AEPSC's role will be both managerial and technical. AEPSC is the program manager. AEPSC and Cook Nuclear Plant personnel will also be involved in all aspects of the technical work including the ongoing, detailed review of work products, with a special focus on analysis and evaluation of study results.

Work for the Cook Nuclear Plant will be conducted in conformance with the appropriate requirements of 10 CFR 50, Appendix B. This will ensure a high degree of quality and allow the PRA results to be utilized to satisfy IPE requirements and assist in future licensing activities, e.g., design change decisions. An independent review team will also be used to provide further confidence in the PRA process and results. The team's efforts will supplement but not be as extensive as the QA reviews.

2.0 Project Methodology

The Cook Nuclear Plant PRA will be performed using standard PRA methods and practices (e.g., event trees, fault trees and fault tree linking) such as those outlined in the PRA Procedures Guide (NUREG/CR-2300). The PRA will also utilize those portions of the IPE Methodology developed by the IDCOR Program that have been proven

to be beneficial for performing a PRA in an efficient manner. Provided below is a description of the Level I, II and III analyses as well as the methodology to be employed for the hydrogen control and external events analyses.

Level I Analysis

The Level I portion of the PRA will consist of the following major tasks:

- Initiating Event Identification
- Plant Data and Information Collection and Analysis
- Dependency Analysis
- Event Tree Development
- Plant Systems Analysis (Fault Tree Development)
- Accident Sequence Quantification

The Cook Nuclear Plant PRA will be a full scope investigation of the plant systems and operator response. The focus will be on performing a realistic assessment of the plant response to potential accident sequences. Models of plant systems will be detailed and explicitly include the performance of key components and equipment. Consideration will be given to using plant-specific component failure data to supplement generic data sources. Where appropriate, Bayesian update techniques of generic data will be used. Fault tree quantification will be performed using the Westinghouse GRAFTER Code System. System success criteria used to determine whether intended safety functions are achieved will be reasonable and not necessarily rely solely on existing success criteria such as those in the UFSAR. Success criteria definition will involve consideration of both system capability and timing. Determination of the proper criteria may be supported by deterministic calculations using appropriate software, such as the Westinghouse TREAT code.

With respect to common cause failure analysis and human error analysis, realistic quantifications will be performed using the Multiple Greek Letter (Beta factor) method, and data from actual plant operations will be reviewed for potential common cause contribution. Detailed analysis of important operator actions will be performed using the THERP technique.

Key operator actions and operator recovery will be explicitly modeled with input from Cook Plant operations. The major proceduralized operator recovery actions will be modeled during the initial quantification effort. Following this quantification, the dominant sequences will be examined in detail to determine if further operator recovery actions should be considered. This approach will ensure that appropriate credit is taken for the operator's role in preventing or mitigating accident sequences.

Level II Analysis

The Level II portion of the PRA will consist of the following major tasks:

- Containment Systems Analysis
- Containment Structural Capability Review
- Containment Event Tree Development
- Source Term Determination

The first task consists of several subtasks that are necessary to provide all the required data for the analysis and the necessary interfaces with the Level I phase of the PRA. Plant specific data will be used and a detailed MAAP parameter file will be assembled. The data collected is consistent with the requirement specified in NUREG-1335 Appendix A. The MAAP code version 3.0B will be used in subsequent analyses.

The dominant accident sequences identified in the Level I phase will be examined to assure that they indeed represent realistic core damage sequences. The plant conditions, available systems, operator actions and duration of systems unavailability will be recorded to assure consistency between the Level I output and the Level II input. Sequences with similar containment response will then be grouped to the same bins of plant damage states.

An understanding of the containment's structural capability is required to assess the extent of protection provided by the containment boundary, to determine the likelihood of containment failure for different accident scenarios and to identify the more probable paths for radionuclide release should containment failure occur. The Cook Nuclear Plant PRA will review the existing plant structural analyses to obtain an understanding of the ultimate capability of the containment. This task will utilize previous experimental work and, if appropriate, studies for other plants. Containment penetrations will also be considered in terms of pressure and temperature capabilities.

The major processes influencing severe accidents and their relations to the Cook Nuclear Plant containment, the plant systems and operator actions will be addressed through the containment event trees (CET). Analogous to a Level I event tree, the CET will logically depict the various containment loads applicable to the Cook Nuclear Plant along with operator and systems response following a severe accident event. End states on the CET will represent different containment responses and potentially different radionuclide release mechanisms and magnitude.

To quantify the CET, severe accident phenomena and their effects on

the Cook Nuclear Plant containment loads will be assessed through a combination of several approaches:

- o Phenomenology position papers,
- o MAAP analysis,
- o Separate effects analysis, and
- o Expert judgement

These are ordered in the priority to be used. Detailed write-ups will be provided for each phenomenon delineated in NUREG-1335. The write-ups will include bounding hand calculations on the containment loads and justification for the appropriate model parameters and assumptions used to assess the containment loads and perform the "best estimate" MAAP analysis. Additional uncertainty analysis with MAAP will be performed to determine the influence of a given phenomenon on the primary system and the containment behavior (e.g., does the physical process in question substantially influence the accident sequence progression and the potential for recovery from the accident state). If a phenomenon is found to have substantial influence on the containment performance, separate effects analyses based on existing data and engineering calculations will be provided.

If phenomena are encountered which have not been addressed through either MAAP analyses or other available experiments, and are judged to be significant, these would be incorporated into the containment evaluation process and expert judgement would be used to assess the impact on the structure and conclusion of the CET analyses. The CET will also include nodes describing lower probability containment response and loads.

This approach for treating the numerous physical processes involved in the severe accident progression and recovery is believed to satisfy the intent of the generic letter and the guidance document (NUREG-1335). Specifically, this:

(1) focuses attention on the phenomena directly affecting the primary system and containment,

(2) produces documentation on the importance of these phenomena for the specific plant and sequence being evaluated, and.

(3) provides a plant specific reference document that can be used for training of utility personnel in severe accident phenomena as suggested in the generic letter.

Furthermore, the approach produces a CET which can be translatable to major accident management decisions/actions. The containment event tree will be applied to each of the plant damage state bins

defined in the first Level II task. The probability of each of the end states will then be determined. Included in this CET quantification process are the pressure and temperatures in the containment for each end state and the timing of key events (e.g., vessel failure and containment failure) as required by NUREG-1335.

Once the CET has been quantified, the likelihood of the end state will be available. The final step in the Level II process will be to estimate the source terms associated with the CET end states. Source terms will be estimated by performing accident sequence analysis and radionuclide transport calculations using the MAAP code. MAAP analyses using the plant specific models will be run for the dominant accident sequence bins. These analyses will predict timing, magnitude, and composition of the release for a particular sequence. When combined with the results of the CET quantification, the probability of experiencing an accident resulting in a given radionuclide release will be determined.

Level III Analysis

The Level III analysis will consist of the following major tasks:

- Development of the Site Model
- Analysis of Offsite Consequences
- Development of Methodology for Estimating Offsite Consequences for a Broad Range of Fission Product Releases

The offsite consequences associated with the fission product release categories identified in the source term analysis will be determined using an appropriate computer code such as MACCS or MIDAS. The exposure as a function of distance from the plant and total population exposure will be determined from the analyses. This will provide the basis for the establishment of the offsite risk profile for the Cook Nuclear Plant.

Hydrogen Control Analysis

To supplement the Level II portion of the PRA and to meet the requirements of 10CFR50.44(c), an analysis will be performed to determine the ability of the hydrogen control system installed in the Cook Nuclear Plant to mitigate the consequences of the release of large amounts of hydrogen into the containment during postulated degraded core accidents. This hydrogen control analysis consists of the following major tasks:

- Base Case Analysis
- Sensitivity Analyses
- Equipment Survivability Evaluation

The analysis will be performed to determine, for accident scenarios resulting in the generation of an amount of hydrogen equivalent to metal-water reaction of up to 75 percent fuel cladding surrounding the active fuel region, whether:

- o The Cook Nuclear Plant hydrogen control system assures containment integrity, and
- o Safety-related equipment operates after being exposed to high temperatures and pressure created by the hydrogen burns.

Current plans are that the Cook Nuclear Plant containment response to hydrogen and steam release during degraded core accidents will be evaluated using the MAAP code. If considered necessary, the CLASIX model developed for the Cook Nuclear Plant in 1985 by Westinghouse may be used on one accident sequence to demonstrate the similarity between the containment response predicted by MAAP and that predicted by CLASIX.

External Event Analyses

For the Cook Nuclear Plant PRA, external event analyses will be performed. The external event analyses will result in either the performance of a new analysis or review of an existing analysis for the following:

- o Plant Specific Seismic Risk Analysis
- o Internal Fire Risk Analysis
- o Analysis of Other External Events, e.g.,
    - External Flooding
    - Wind Analysis
    - Hazardous Material Transportation
    - Aircraft Hazards
    - Other Transportation Hazards
    - Turbine Missiles
    - Other Hazardous Materials at the Cook Nuclear Plant Site

Results of the external event analyses will be used as input to the PRA in defining initiating events, in developing event and fault trees for accident sequence and system analysis, and in quantifying accident sequences. In this manner, the resultant PRA product will provide a comprehensive assessment of risk for comparison with NRC and industry safety goals.

3.0 Project Schedule

The development of the Cook Nuclear Plant PRA and IPE submittal will be completed within the time allotted by Generic Letter 88-20. Specifically, the Level I portion of the PRA project is currently scheduled to be completed in approximately 16 months. The Level II portion of the PRA Project will overlap the performance of the Level I effort and is expected to be completed approximately one month afterwards. The Level III effort and the Hydrogen Control Analysis will be completed following the Level II effort. Most of the external event analysis will be performed in parallel with the Level I PRA effort with the remainder scheduled to be completed within the 24 month estimated total project time period.

Figure 1 presents an overview of the current project schedule showing the relative timing of the primary tasks. The schedule allows for sufficient float for unanticipated adjustments without compromising the 3-year limit for submitting the IPE results to the NRC as set forth in Generic Letter 88-20.

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                             1989              1990                       1991

                         8 9 10 11 12 1 2 3 4 5 6 7 8 9 10 11 12 1 2 3 4 5 6 7 8 9 10 11 12

LEVEL I PRA              XXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXX

LEVEL II PRA               XXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXX

LEVEL III PRA                                                    XXXXXXXXXXXX

HYDROGEN CONTROL ANALYSES                                      XXXXXXXXXXX

EXTERNAL EVENT ANALYSIS                XXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXX

IPE SUBMITTAL TO NRC                                                                  XX
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Figure 1

COOK NUCLEAR PLANT PRA PROJECT CURRENT SCHEDULE