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Author(s): John F. Carew and David J. Diamond

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Responsible NRC Individual and NRC Office or Division: Mr. Seymour Weiss
Reactor Safety Branch
Div. of Operating Reactors
U.S. Nuclear Regulatory Comm.
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Brookhaven National Laboratory
Upton, NY 11973
Associated Universities, Inc.
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REVIEW OF MODIFICATIONS PERFORMED TO CORE MONITORING SYSTEMS RELATED TO CORE RELOADING

John F. Carew

David J. Diamond

Reactor Core Safety Analysis Group
Thermal Reactor Safety Division
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

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ABSTRACT

The recent increase in the number of licensees selecting new fuel suppliers for reload cycles has resulted in a trend toward Core Monitoring Systems (CMS) for which the cycle dependent data and the CMS software are supplied by different vendors.

At the request of and under the direction of the Division of Operating Reactors, USNRC, a review of the qualification and documentation for these CMS has been made. Several potential problem areas in the determination of CMS cycle dependent input, involving empirical normalizations and relatively complex neutronic analysis, were identified. As representative of present qualification and documentation practices, Yankee Atomic Electric Co., Virginia Electric Power Co., Nuclear Associations International, Exxon Nuclear Co., Northeast Utilities Service Co. and Jersey Central Power and Light were selected and reviewed in detail.

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I. Introduction

At the request and under the direction of the Division of Operating Reactors, USNRC, a review of modifications to Core Monitoring Systems (CMS), associated with core reloading has been performed. The purpose was to provide assurance that as new fuel vendors or the licensees themselves provide cycle dependent process computer modifications, instead of the original fuel vendor who provided the process computer program, the plant s[illegible]ns are not significantly reduced. Specifically, this task was [illegible]censee procedures used to establish the accuracy of cycle de[illegible]ogram modification for the CMS. This report document[illegible]iew of the qualification and documentation practices.

Typically, the prog[illegible]ion of cycle dependent input data for the core monito[illegible]ng the initial cycle of an operating reactor is performed by the NSSS vendor. However, in subsequent cycles some licensees have selected new fuel vendors and for these cases the CMS cycle dependent data and CMS software may be supplied by different organizations. In addition, in recent years sophisticated nuclear design codes have become available and some licensees are performing, or are planning to perform, their own nuclear safety analysis including the calculation of the CMS cycle dependent input data. Also, in order to improve their CMS, licensees are incorporating modifications to the program itself. This software/data supplier change in many cases involves proprietary issues and it is not clear that the new fuel supplier has access to a description of the CMS that was provided by the original fuel supplier.

Implicit in this review is the assumption that the cycle dependent data generation, transmission and loading, and software modification and loading are performed under the same Nuclear Reactor Regulation and Inspection and Enforcement (I & E) quality assurance and auditing programs, regardless of whether or not the fuel vendor was changed. This review then concentrates on the licensee qualification of (1) the methods used to determine the CMS cycle dependent data and (2) the software used to calculate the operating core performance. As a sample of industry practices, the following data methods and/or software suppliers were selected; Yankee Atomic Electric Co. (YAEC), Virginia Electric Power Co. (VEPCO), Nuclear Associations International (NAI), Exxon Nuclear Co., Northeast Utilities Services Co. (NUSCO) and Jersey Central Power and Light (JCPL). A review of the qualification procedures employed by these organizations was then performed. This involved meetings in Bethesda and at licensee facilities as well as extensive telephone conversations.

As a first step, a study was performed to identify potential problem areas in the calculation of CMS input data and qualification methods, which could be particularly acute in the case of a new fuel vendor.

Although problems are known to exist in the in-core instrumentation,*

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*Typical examples being; detector mislocation, failure and anomalous sensitivity decay.

these are independent of the software/data supplier problem and will not be considered here.

II. CMS Data and Qualification

A. Input Data

The CMS input data is precalculated by a set of relatively complex nuclear and thermal-hydraulic programs. Typically, the data associated with the interpretation of the in-core detector signals results from detailed few-group neutron diffusion theory analyses. The thermal-hydraulic data (generally required only for BWRs) is determined by multi-channel core thermal-hydraulic calculations. An important requirement of this data and the underlying analyses is that it must span, to within accepted accuracy, all states realized during the cycle of interest. The following summarizes specific difficulties involved in meeting this requirement.

1. Detector Sensitivity. As exposure accumulates on the fixed in-core detectors the flux sensitivity decreases due to depletion of the neutron sensitive material (typically ^{235}U for fission detectors and rhodium or vanadium for self-powered detectors). Since this sensitivity loss is not uniform, with the high (low) powered nodes receiving the largest (smallest) decrease in sensitivity, this effect must be accounted for in the CMS. This is especially true for those B & W and CE plants without movable in-core systems to calibrate the fixed in-cores. The sensitivity decay rate is generally difficult to determine and varies significantly during the life of the detector.

2. Detector Space-Energy Representation. In the neighborhood of the detector, the thermal neutron flux undergoes rapid spatial variation due to (1) moderator thermalization and (2) flux perturbation due to the presence of the detector. In general, this behavior requires a detailed spatial transport calculation in order to avoid errors in the calculated instrument response.

3. Boundary Calculations. The steepest flux gradients generally occur near the radial and axial core boundaries. In order to interpret the in-core signals in these locations, detailed diffusion theory analyses including explicit boundary representations are required. These analyses are further complicated in the case of BWRs, since the signal-to-power correlation depends on the local control rod pattern at the radial boundary. In the CE and B & W plants without movable in-cores an axial boundary condition is required which also depends on control density.

4. Asymmetric Operation. For B & W and GE plants, in the case of an asymmetric control rod pattern the CMS operates in an asymmetric mode. In this situation the instrument density is reduced by a factor of ~ 4 in the region of asymmetry and the interpolation of in-core signals becomes less accurate. Additional cycle-dependent input data are required to determine the core power distribution and since the corrections depend on the specific asymmetry, it is difficult to precalculate.

5. Control-History Effects. In general, the signal-to-power correlation used to interpret in-core signals depends not only on the instantaneous control pattern but also on the recent control-history of that assembly. This dependence is difficult to calculate and especially difficult to incorporate in the CMS software.

6. Reload Cores. A reload core includes both fresh and exposed fuel typically arranged in a heterogeneous pattern with significant exposure variation. Since the input data required to correlate signal to local power is exposure dependent, this variation results in a significant increase in the number of calculations required to track the detailed cycle exposure dependence. Also, the form of exposure dependence used for the correlations in the CMS libraries is fuel dependent and becomes very important in this case. In reload cores the number of distinct fuel assembly types is greater than for initial cycles, and the CMS libraries become proportionately more complex.*

7. Axial Interpolation of Fixed In-Cores. Both B & W and CE CMS perform an axial interpolation of the fixed in-core measurements to determine the assembly axial power distribution. (GE, W and recent CE systems include movable in-cores to provide a continuous axial trace.) The constants used in this interpolation may depend significantly on the fuel design and loading, although the present CMS do not take this into account.

8. Channel Friction Factors. Channel friction factors are used in the BWR CMS analysis to relate bundle pressure drops and flows. In practice, these constants are used to normalize the simplified CMS thermal-hydraulic model to more detailed core thermal-hydraulic calculations. Significant uncertainties may be introduced if these factors are not determined accurately.

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*Since the W in-core libraries do not recognize identical assemblies, initial and reload libraries are similar.

9. Empirical Data. In addition to input data precalculated from detailed nuclear and thermal-hydraulic analysis, certain input data is determined by normalizing the CMS results to (1) associated 3-D calculations, (2) experimental data and (3) operating data. The resulting empirical data is generally based on comparisons over many cycles and requires frequent verification. Examples of this are the radial and axial boundary corrections, detector sensitivity decay constants and the adjustments made to the weighting of the neighboring detectors in determining the power in unmonitored assemblies.

In addition to the above technical problems, the calculation of the CMS input generally requires input from the nuclear and thermal-hydraulic designers, instrumentation and systems groups and computer programmers.* This calculation process is further complicated when these groups belong to different vendor organizations. For example, at Kewanee, Exxon will supply the fuel, W supplies the software, NAI will supply the nuclear methodology and WPSC will generate the data. In many cases, design information is covered by proprietary agreements and is difficult to obtain. Also, the standard vendor software packages (CE-INCA,[1] W-INCORE,[2] B & W-NASP,[3] and GE-Process Computer[4]) are generally proprietary and in many cases may only be executed by the utilities.

B. General Qualification

The CMS is used to monitor the Limiting Conditions of Operation (LCO) as specified in the Technical Specifications (either directly as in the case of BWRs and many PWRs or indirectly in order to calibrate fixed in-core detector

*Typical required data includes: fuel, lattice, core, orificing, spacer, bypass and core flow design data; temperature, flow, and flux instrumentation data.

outputs). For this safety application, the CMS requires a complete review and qualification. In addition, the LCOs include a specific margin for monitoring uncertainties and these may only be determined by a detailed qualification of the CMS analysis. At present, Topical reports have been submitted by B & W,* CE, W and GE for their "standard" core monitoring systems specifying the monitoring uncertainties. The qualification documentation submitted to date, for the non-standard systems is limited.

In order to qualify the CMS, an estimate of the monitoring uncertainties must be made. This is generally done by comparing CMS results with either an independent calculational or experimental benchmark. The calculational benchmark is more convenient since the results are generally available as part of the Core Follow Program, and may be made over many operating states and cycles. In fact, these comparisons are required as part of the PWR start-up physics tests. BWRs are presently not required to make this comparison. It is important to make comparisons over many cycles in order to properly account for instrumentation and system degradation, exposure heterogeneities, etc. It is also important to note that, in practice, this calculational benchmark is usually not completely independent due to frequent inter-normalizations of the CMS and calculational model (empirical normalizations). On the other hand, only a few experimental benchmarks exist and in most cases are not available or applicable to the licensee's system. Consequently, qualification of the non-standard systems has been limited to the calculational benchmarks.

*The proprietary version of the B & W qualification topical (B & W 10120) is expected to be submitted July, 1978.

III. Qualification Practices

The following is a summary of an informal review of available qualification and related documentation from the selected utilities, nuclear consultants and fuel manufacturers.

A. Virginia Electric Power Co. (VEPCO). VEPCO has developed a PDQ07 nuclear analysis system to support their start-up and cycle operation of the Surry 1 and 2 and North Anna (W) reactors. In addition to reactor physics and fuel management analysis, this system will be used to generate input for the INCORE system. As qualification for this system, VEPCO has submitted a topical report[5] presenting ~ 70 detailed comparisons of measured and predicted core power distributions over the first two cycles of Surry 1 and 2. Comparisons of assembly-wise power distributions and $F^N_{\Delta h}$ are presented for the VEPCO and vendor models at various power levels, bank positions and burnup. (All INCORE data was determined using the VEPCO PDQ model.) These comparisons indicate the VEPCO system measurement and calculational uncertainties are to within ~ 2%* on the assembly average power distribution and ~ 2.5% on the peak rod $F^N_{\Delta h}$. These values are consistent with the vendor model and with the assumed W uncertainty analysis. It is noteworthy that this qualification was submitted prior to the application of the model to the Surry and North Anna plants.

In addition, VEPCO has provided the NRC with extensive measurement/predicted power distribution comparisons for the Surry nuclear station periodically throughout

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*All percent differences are understood to be at the one-sigma confidence level.

the present cycle as further demonstration of the validity of their nuclear analysis model.

B. Yankee Atomic Electric Co. (YAEC). YAEC has developed a nuclear analysis system to provide reactor physics and core follow support to Maine Yankee and Yankee Rowe. Specifically, this model is used to determine the Maine Yankee INCA and Yankee Rowe INCORE CMS input data. The qualification of this nuclear analysis system for application to Main Yankee is provided in a recent YAEC topical.[6] While detailed comparisons of the YAEC-calculated and CE-INCA (i.e., using CE input data) results are included, no YAEC-INCA results are presented. (The definition of the INCA data and Maine Yankee cycle-3 values are included in Reference-7).

Prior to cycle-3 startup, no qualification of the YAEC-INCA methods were submitted for Maine Yankee. As part of the startup physics test, comparisons of predicted and YAEC-INCA results were required and do provide a valid (although limited) qualification of the YAEC methods. In addition, as part of their core follow program, YAEC performs monthly comparisons of calculation and the YAEC-INCA and INCORE measurements. In response to the present review, a set of 8 PDQ/INCORE radial power maps for Maine Yankee Cycle-3 were submitted informally. These comparisons indicate an ∿ 2-3% discrepancy in assembly power and are in agreement with the 2.5% CE uncertainty value.

C. Nuclear Associations International (NAI). NAI has developed the Advanced Recycle Methodology Program (ARMP)[8] as a general purpose nuclear analysis system for EPRI. This program will provide the licensees the capability to perform their own nuclear analysis and specifically, a large part of the

methodology required to determine CMS input data. In fact, at present a few PWR licensees are determining their CMS (INCORE, INCA) input using this system. In support of their system, NAI conducts seminars on model construction and performs audits of licensee's application.

No formal qualification of the NAI CMS input methodology has yet been performed, however, a limited number of comparisons of the ARMP-predicted and measured instrument signals have been made. No documentation of these comparisons is available. NAI is presently preparing (under contract to EPRI) a detailed uncertainty and qualification evaluation for its CMS input methodology for the W-INCORE and GE CMS.

D. Exxon. As a nuclear fuel supplier, Exxon provides CMS input data to each of its customers unable to perform the calculations himself. In fact, Exxon has supplied this data for (1) the INCA system at Palisades and in the near future for Maine Yankee, (2) the INCORE system at H. B. Robinson-2 and Ginna, (3) the DETECTOR* system at D. C. Cook-1 and (4) the JCPL CMS† at Oyster Creek.

As a limited qualification of their data generation methods Exxon has submitted a set of ($\sim$ 10) comparisons of measured and predicted radial instrument signals for D. C. Cook-1 (cycles 1 and 2) and H. B. Robinson (cycles 5 and 7) at various exposures. These radial maps indicate an $\sim$ 1-4% difference which is comparable to the 2.5% values used by CE and W.

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*DETECTOR is a CMS code written by the American Electric Power Co. for use at D. C. Cook.

†JCPL has written their own CMS code for use at Oyster Creek.

Also, in the recently submitted Supplement-2 of the Exxon Nuclear Design Methods Topical[9] PDQ/CMS comparisons of the radial power distribution for H. B. Robinson (cycles 4 and 5), D. C. Cook-1 (cycle-2) and Palisades (cycle-2) are presented for various exposures and rod insertions. Although these comparisons are presented as a basis for the PDQ model, they also may be interpreted as qualification for the Exxon CMS data methods. The Cook results (3-maps) agree to within 2-4%, the H. B. Robinson data (6-maps) agree to within 1.5-3%, and the Palisades results (2-maps) agree to within 3-4%. It should be noted, however, that although these differences are in reasonable agreement with the PWR vendor CMS uncertainties, they are based on a relatively small set of comparisons having limited documentation at best.

Exxon is presently preparing a Topical Report (~ Fall '78 submittal) including detailed qualification of their CMS input methods. This evaluation will include an overall uncertainty analysis, as well as an identification of the various component uncertainties.

E. Northeast Utilities Services Co. (NUSCO). NUSCO performs the reactor physics, fuel management and core follow analysis for the Connecticut Yankee and Millstone 1 and 2 plants.

The Millstone-1 CMS software and input data are both supplied by GE, the original NSSS vendor.

Millstone-2 is a CE reactor for which CE supplies the fuel and INCA input data. CE provided the original version of INCA which NUSCO implemented on their

computer. NUSCO has full responsibility for the CMS software and makes modelling changes when appropriate. Examples of these are the new CE rhodium detector burnup correlation, axial coupling model and pin power exposure correlation. For the most part these changes are defined by CE. In addition, NUSCO has improved and simplified many of the INCA editing routines. No additional documentation or qualification has been submitted in support of these changes. It is noteworthy that, in fact, many licensees have introduced similar changes into their CMS's without submitting any supporting documentation or qualification. (Although not explicitly identified, the software and requalification requirements should follow from Reg. Guide 1.33, "Quality Assurance Program Requirements (Operation)".)

Except for INCA modelling changes, the original CE qualification should apply to Millstone-2. In addition, NUSCO performs periodic INCA/simulator comparisons as part of their core follow program. A few typical power distribution comparisons were presented at a meeting at NUSCO (May, 19, 1978). These comparisons indicated an $\sim$ 2% discrepancy with calculation in good agreement with the CE uncertainties. NUSCO also has access to the CE time-share version of INCA, CECORE, which it uses to evaluate the Millstone-2 INCA system.

It should be noted that in the near future NUSCO intends to purchase fuel from W for the Millstone-2 plant which will create a software (CE)-data (W) mismatch.

Connecticut Yankee is a W reactor which NUSCO monitors with the W INCORE system. NUSCO implemented the original version of INCORE, has full software responsibility and has made many editing changes. Since B & W supplies the fuel and INCORE data for Connecticut Yankee, a software-data mismatch exists.

No formal qualification for this system has been submitted. However, in support of the B & W-INCORE, NUSCO presented selected PDQ/INCORE power distribution maps at the (May 19, 1978) meeting at NUSCO. These comparisons indicated an ~ 2% agreement, consistent with the W uncertainties.

F. Jersey Central Power and Light (JCPL). JCPL operates the Oyster Creek BWR/2 plant. The CMS used to monitor this reactor was designed and implemented by JCPL and, since the fuel and CMS input data is supplied by Exxon, a mismatch exists.

No formal documentation or qualification of this system has yet been submitted, although several I & E inspections have been conducted. As part of their on-going qualification, JCPL makes periodic CMS/simulator comparisons, however, this data is not available.

IV. Summary and Conclusions

The CMS is generally used to calibrate excore detectors and/or perform the determination of the LCO's as specified in the Technical Specifications. As a safety related system, it therefore requires a detailed qualification, especially in situations where a CMS software/data mismatch exists.

The major conclusions of the review of the CMS qualification in this situation are as follows:

1. The number of changes from the original supplier of CMS input data is expected to increase significantly in the near future.
2. CMS software changes are introduced by the licensees without submitting additional documentation or qualification.

3. A minimum of qualification is performed before startup. At startup, PWR's perform a "one-state" qualification test comparing CMS and predicted results. Typically, BWR's do not make this startup qualification test.

4. Essentially all licensees conduct a continuous (~ monthly), qualification program in which CMS and predicted results are compared. This qualification is not completely valid, however, due to internormalization of the CMS and prediction models.

5. Essentially no formal quali[illegible] as been submitted in support of these non-standard systems.

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