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402/636-2000

September 1, 1994
LIC-94-0169

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555-0001

References:
1. Docket No. 50-285
2. Letter from OPPD (W. C. Jones) to NRC (R. A. Clark) dated December 3, 1982 (LIC-82-389)
3. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated September 24, 1993 (LIC-93-0135)
4. Letter from NRC (S. D. Bloom) to OPPD (T. L. Patterson) dated February 17, 1994
5. Letter from NRC (J. E. Richardson) to CEOG (J. J. Hutchinson) dated April 12, 1993
6. ABB Combustion Engineering Licensing Topical Report No. CEN-415, Revision 1-A, "Modification of Post Accident Sampling System Requirements," dated September 1993

Gentlemen:

SUBJECT: Revision of Post Accident Sampling System (PASS) Commitment for Measurement of Reactor Coolant pH (TAC No. M88079)

By this letter, Omaha Public Power District (OPPD) notifies the NRC of a revision to a PASS licensing basis commitment related to measurement of reactor coolant pH at Fort Calhoun Station (FCS). This revision will delete the requirement for measurement of reactor coolant pH, by taking credit for the trisodium phosphate baskets installed in the containment on the basement level.

NUREG-0737, Section II.B.3 contains no specific requirement to monitor reactor coolant system pH. However, measurement of reactor coolant and containment sump pH is recommended by Regulatory Guide 1.97. As part of Reference 2, OPPD committed to certain PASS pH analysis accuracy values. In Reference 3, OPPD revised these values. The NRC approved these revised values in the Safety Evaluation which was transmitted by Reference 4. To assure compliance with applicable PASS commitments, OPPD has been periodically measuring the pH of the reactor coolant and safety injection systems through performance of procedures CH-SMP-PA-0001, "Post Accident Sampling System Normal Operation," and CH-SMP-PA-0002, "Post Accident Sampling System Accident Operation."

OPPD intends to delete from the PASS program the requirement to measure pH of the reactor coolant and safety injection systems. The procedures noted above will be revised accordingly pursuant to 10 CFR 50.59. The justification for this change is that post accident pH is controlled by the passive additive delivery of trisodium phosphate dodecahydrate (TSP) via the storage baskets in the containment on the basement level.

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As described in the Updated Safety Analysis Report for FCS (Section 4.4.3), the TSP storage baskets contain at least 40 cubic feet of TSP which have been analyzed to raise the pH of the sump solution to ≥7. Operability of the TSP baskets is demonstrated through performance of surveillance procedure CH-ST-CH-0002, "Phosphate Basket Inspection," on a refueling frequency to satisfy Technical Specification 3.6(2)d.

The preceding justification is within the scope of Section 3.1, *RCS pH*, of the attached Reference 6 topical report. Because NRC approval of Section 3.1 of the topical report is documented by the NRC letter (Reference 5) incorporated into the report, OPPD considers the aforementioned licensing basis commitment change to be acceptable.

If you should have any questions, please contact me.

Sincerely,

W. G. Gates
for Vice President

WGG/tcm

Attachment

c: LeBoeuf, Lamb, Greene & MacRae (w/o attachment)
L. J. Callan, NRC Regional Administrator, Region IV (w/o attachment)
R. P. Mullikin, NRC Senior Resident Inspector (w/o attachment)
S. D. Bloom, NRC Project Manager

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

APR 12 1993

Mr. John J. Hutchinson
Florida Power and Light
700 Universe Boulevard
Juno Beach, FL 33408

Dear Mr. Hutchinson:

SUBJECT: APPROVAL FOR REFERENCING OF LICENSING TOPICAL REPORT NO. CEN-415, MODIFICATION OF POST ACCIDENT SAMPLING SYSTEM REQUIREMENTS, REVISION 1, TAC M82498

We have completed our review of the subject topical report. We find that only a part of the material presented in this report could be approved for referencing in licensee applications to the extent specified and under limitations delineated in the report and associated NRC evaluation which is enclosed. The evaluation defines the basis for the limited approval of the report. The following topics in the report were approved: (1) measurement of pH of reactor coolant, (2) measurement of containment hydrogen concentration, (3) heat tracing of sample lines, (4) oxygen analysis of reactor coolant, and (5) sample points requirements. Two topics were found not to meet the guidelines in Section II.B.3 of NUREG-0737 and, therefore, were not approved. PASS sampling has to meet the three hours time limit and measurement of hydrogen/total gas in the reactor coolant should constitute the primary method for determining the conditions inside the reactor vessel. However, gas sampling could be performed either by PASS or a normal sampling system, if the General Design Criterion 19 on the radiation exposure could be met.

We do not intend to repeat our review of the approved matters described in the report when the report appears as a reference in the license applications except to assure that the material presented is applicable to the specified plant involved. Our approval applies only to matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that the Combustion Engineering Owners Group (CEOG) publish approved versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted version should incorporate this letter and the enclosed evaluation between the title page and the abstract.

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If the Commission regulations or staff implementation guidelines change such that our conclusions on the acceptability of the report are modified, CEOG and/or the licensees referencing this topical report will be expected to revise and submit their respective documentation, or submit justification for the continued effective applicability of the topical report without revisions of their respective documentation.

Sincerely,

J E Richardson

James E. Richardson, Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosures:
As stated

# SAFETY EVALUATION REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO MODIFICATION OF POST ACCIDENT SAMPLING SYSTEM REQUIREMENTS MATERIALS AND CHEMICAL ENGINEERING BRANCH TAC NO. M82498

## 1.0 INTRODUCTION

The objective of the NRC staff review of the report in Reference 1 is to evaluate the proposed clarifications and modifications and, if appropriate, to approve the departures from the current NRC requirements.

The initial NRC requirements for the Post-Accident Sampling System (PASS) were specified in Reference 2. However, some of them were subsequently revised or are in the process of being revised. The modifications to the original requirements in Reference 2 are described in References 3 and 4.

The report in Reference 1 contains the following clarifications and modifications of the existing requirements:

- Clarification of the requirement for measurement of sump pH.
- Deletion of the requirement to measure containment hydrogen and relying on the safety-grade containment hydrogen monitors.
- Clarification of the requirement for heat tracing of the containment atmosphere sample lines and modification of the Core Damage Assessment Procedure to base the core damage predictions on noble gas concentrations.
- Use of hydrogen/total gas measurements in the reactor coolant samples only as a backup method to the Reactor Vessel Level Monitoring System and include an option for using either PASS or the normal sampling system for performing these measurements.
- Clarification of the requirements for oxygen analysis in the reactor coolant.
- Clarification of the requirement for sample points in PASS.
- Modification of the time limit for the determination of dissolved gases, activity and boron concentrations in the reactor coolant.

## 2.0 EVALUATION

The above listed clarifications and modifications were reviewed and evaluated by the NRC staff, relative to the criteria specified in the referenced documents and the current NRC staff interpretation of these criteria. The results of these evaluations are presented below:

2.1 Measurement of pH of Reactor Coolant

There is no specific requirement in the PASS specifications for measuring pH in the sump. However, Criterion 10 in Reference 2 implies that appropriate parameters should be measured in order to describe radiological and chemical status of the reactor coolant system. Also, for the same reasons, Regulatory Guide 1.97 recommends to measure pH of the primary coolant and in the containment sump. The justification provided in the report (Reference 1) for deleting of pH measurement is based on the fact that in the post-accident environment pH in the sump is controlled at a value of ≥7 either by the addition of sodium hydroxide to the containment spray solution or by a passive pH control achieved by buffering action of the chemicals stored in the sump (usually trisodium phosphate dodecahydrate). The reliability of this control is sufficiently high to ensure that neither corrosion of reactor components nor reevolution of dissolved iodine will occur. In view of the fact that the current specifications for PASS do not specifically require pH measurement in the sump water, and the existence of the reliable methods for pH control, we consider the proposed deletion of pH measurement in the sump water is acceptable.

2.2 Measurement of Containment Hydrogen Concentration

Criteria 2b and 8 in Reference 2 require PASS to have a capability to measure hydrogen levels in the containment and, if inline monitoring instrumentation is used, to provide a grab samples backup. However, the requirement in Reference 2 specifies that hydrogen in the containment be monitored by the safety-grade hydrogen monitors which would be capable to provide hydrogen concentrations after an accident. In the report (Reference 1) it is requested, therefore, that the requirement for containment hydrogen measurement by PASS be deleted. The staff concludes that because the hydrogen monitors in the containment provide adequate capability for monitoring post-accident hydrogen, there is acceptable justification for this deletion and finds the requested modification acceptable.

2.3 Heat Tracing of Sample Lines

The purpose of heat tracing of PASS sample lines is to prevent iodine plateout which would result in incorrect measurements of iodine concentration in the containment and could constitute an important source of errors, if iodine concentrations were used for assessing degree of core degradation. Criterion 11 in Reference 2 specifies, therefore, that provisions should be made for reducing this plateout. However, if iodine is not used for determination of the degree of core degradation, the information about its concentration is not needed and its plateout in the PASS sample lines does not constitute a problem. This issue was discussed in Reference 3, and it was determined that when the core damage assessment procedures are based on representative containment atmosphere xenon and or krypton activities, there is no need to sample iodine in order to meet the requirements of Reference 2. The

modification proposed in the report (Reference 1) would modify the Core Damage Assessment procedure for the CE plants to base the assessment on noble gas concentrations. This modification is consistent with the current NRC policy and the staff find it, therefore, acceptable.

2.4 Measurement of Hydrogen/Total Gas in Reactor Coolant

Measurement of hydrogen or total gas in the coolant water is required so that the operators would know about potential problems which may occur, if during a depressurization dissolved gas is released into the reactor vessel and forms a bubble which could interfere with heat removal by natural circulation. Also, since most of the hydrogen in the reactor vessel is formed by oxidation of fuel cladding by steam, information on the amount of hydrogen dissolved can give some indication on the degree of core damage. Criterion 2c in Reference 2 requires that PASS should have a capability to provide within 3-hour time frame quantitative determination of dissolved gases in the reactor coolant, and Criterion 4 specifies that this measurement should not be necessarily performed on pressurized sample, if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. In addition, Criterion 4 states that measurement of either total dissolved gases or hydrogen gas is considered adequate.

The modification proposed in the report (Reference 1) consists of replacing dissolved gas measurements by PASS as a primary method for monitoring gas release in the reactor vessel by the procedures relying on the Reactor Vessel Level Monitoring System and the Emergency Procedures for maintaining natural circulation in the reactor vessel. A core damage assessment will be performed using coolant radionuclide concentration whose determination is required by Criterion 2a in Reference 2. Dissolved gas determination by PASS will be used only as a confirmatory method and as such would not be needed to meet the stringent requirements specified in Reference 2. Dissolved hydrogen could be measured by an inline hydrogen monitor for which dissolved gas results are easily obtainable. The grab samples backup will be of a secondary importance and will not be subject to 3-hour time requirements for taking the first sample. Also, there will be an option of using either PASS or normal process sampling system for performing this measurement, depending on activity levels.

The report claims that using the process sampling system would improve the accuracy of measurements of low concentrations of dissolved gases, because the inherently small volumes of PASS samples make this determination very difficult. The report (Reference 1) claims that the requirements in Reference 2 are applicable to the plants similar to TMI-2 in design. In the existing CE plants, the information for which gas concentration in reactor coolant is needed could be obtained by alternative means. Combustion Engineering has developed the Emergency Planning Procedure to assist the licensees in developing plant-specific procedures to account for various accident scenarios, including recovery from loss of natural circulation.

The staff considers concentration of dissolved gases in the reactor coolant to be one of the most direct parameter in diagnosing of the problems related to the interruption of natural circulation caused by release of noncondensible gases in the reactor vessel. It also constitutes an important parameter in determining the degree of core damage from the estimate of the amount of fuel cladding oxidation. Although, as suggested in the report, other sources could be used for obtaining this information, the direct determination of the amount of dissolved gas in the reactor coolant still provides the most reliable evidence of the potential problems caused by the presence of dissolved gases in the reactors vessel. This is especially applicable to some postulated accident sequences in which the reactor coolant system is intact at reduced pressure, and heat is removed by subcooled decay heat removal. For these cases, it will not be possible to evaluate concentrations of the dissolved gases in reactor coolant by measuring their concentrations in the containment by the inline hydrogen monitor. For PWRs exposed to these conditions, information on the amounts of dissolved hydrogen in the reactor coolant is an important factor in evaluating post-accident conditions in the reactor vessel. Therefore, the staff concludes that the modification proposed in the report is not acceptable and PASS in the Combustion Engineering plants should meet the requirements of Criterion 2c of Reference 1. However, the option for using either PASS or the normal sampling system for taking coolant samples is acceptable provided the radiation exposure limits of General Design Criterion 19 are not exceeded.

2.5 Oxygen Analysis of Reactor Coolant

There is no specific requirement in Reference 2 for measuring concentration of oxygen in the reactor coolant, although Criterion 4 of this reference recommends that oxygen concentration be determined. The change proposed in the report (Reference 1) deletes the requirements for oxygen analysis and instead specifies that it be obtained by the calculation based on the oxygen concentrations in the Reactor Water Tank or containment atmosphere. Since Reference 2 does not mandate for PASS to have the capability for measuring oxygen concentration in the reactor coolant, and there is a provision for obtaining this information by indirect means, the change proposed in the report is acceptable.

2.6 Sample Points Requirement

Specifications concerning sampling by PASS are contained in Criteria 1 and 11a of Reference 2. Criterion 1 requires for PASS to have a capability for obtaining reactor coolant and containment atmosphere samples and Criterion 11a specifies that these samples be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The report

(Reference 1) clarified this requirement stating that the required sample points in PASS are only those needed for taking containment atmosphere and reactor coolant samples. This statement does not contradict the requirements of Reference 2 and is, therefore, acceptable.

## 2.7 Sampling Time Requirement

Criteria 1 and 2 in Reference 2 specify that PASS should have a capability to obtain and analyze reactor coolant and containment samples within three hours from the time a decision is made to take a sample. As the decision could be made anytime during the accident management phase, PASS should be designed for obtaining analytical results within three hours from the beginning of the accident. Reference 3 provides clarification that the three hour time limit is not a rigorous requirement and a small departure from this value is permissible. Also, the specified time period applies to the first sample only and it does not include the time required for sampling preparation (i.e., suiting up, radiation measurements, etc.).

In the report (Reference 1) a statement is made that the time requirement is only a recommendation applying to the first sample only. The statement implies that the licensee is at liberty to provide PASS sampling results at any time during the accident. This was not the intention of the clarification in Reference 3. A specified time limit for providing PASS sampling results is still a valid requirement. However, the NRC is in process of revising some of the time limits specified in Reference 2. For PWRs, the new time limits for taking the samples and analyzing them for dissolved gas and activity could be extended to 24 hours, and for boron to 8 hours, after the end of power operation. Should the Commission approve this revision, the change in sampling times stated above would be acceptable.

The reason for the proposed extensions is the availability of other sources of post accident information which may satisfy immediate needs of the operators.

In the case of dissolved gas analysis, the PASS results are needed to provide only a confirmatory evidence on the potential for formation of bubbles of noncondensible gases in the reactor vessel that could interfere with core cooling. The information on core cooling could be obtained by other instrumentation, e.g. reactor vessel water level, which although less reliable, could provide, in a much shorter time, an estimate of the thermal-hydraulic conditions in the reactor vessel. Similarly, the analyses of PASS water samples for activity are needed only to confirm the information on the degree of core degradation obtainable by other means, such as high-range containment radiation monitors, or core-exit thermocouple readings. These instruments give the operators usually enough information to manage the accident in its initial stage.

Information on accurate boron concentration in the reactor coolant is essential in preventing criticality during the degraded core accident. It is needed relatively soon (during accident management phase) to verify the operators' estimates of boron concentration that are based on mixing ratios. After reactor isolation this information could not be obtained from the process sampling system and PASS becomes the only means of providing direct boron concentration. However, for the plants with the neutron flux instrumentation complying with the Category 1 criteria of Regulatory Guide 1.97, i.e. having fully qualified, redundant channels that have capability to monitor the power range of 10E-6 percent to full power, enough information could be gained from the instrumentation, and for these plants there is a good justification for postponing measurement of boron concentration to 8 hours after the end of power operation.

In view of considerations discussed above, the modifications to the time requirements proposed in the report (Reference 1) are not acceptable. The time requirements for making dissolved gases, activity and boron concentration determinations should remain unchanged from being those specified in Reference 2. However, as discussed in Reference 3, slight departure from the specified time requirements is permissible. Future revision and relaxation of these requirements would have to be approved by the NRC.

## 3.0 SUMMARY AND CONCLUSIONS

The intentions of the clarification and modifications presented in this topical report is to identify alternative methods of meeting the intent of the requirements in Reference 2. In general, the proposed changes represent departure from using PASS as the primary means of obtaining the post accident information on reactor coolant and containment atmosphere and relying on the information provided by other safety grade equipment. The report provides clarifications in the following areas: pH measurement of reactor coolant, heat tracing of sample lines, oxygen analysis of reactor coolant and sample points requirement. These clarifications provide an interpretation of the requirements presented in Reference 2. They do not depart from these requirements and are, therefore, acceptable. On the other hand, the modifications proposed in the report (Reference 1) are not acceptable because they include the changes which in two cases significantly alter the intent of the original requirements.

The modification consisting of deletion of measurement of hydrogen concentration in the containment was justified by existence of containment monitoring system and it was accepted by the staff in the previous licensing action.

Use of hydrogen/total gas measurements as a backup to other methods for determining conditions in the reactor vessel was found not acceptable because reliable estimate of gas release in the reactor vessel could be provided only by direct measurements of dissolved gas concentration. However, it is acceptable to take coolants samples using normal sampling system, provided radiation exposure requirements are met.

The modification to consider time requirements for performing PASS analyses as only a recommendation which would allow PASS sampling results to be obtained any time during the accident is not acceptable. The time requirement for taking first sample should be in conformance with the requirements specified in Reference 2. However, in the future, subject to the Commission approval, this requirement may undergo some changes.

## 4.0 REFERENCES


1) Report CEN-415, Revision 1, "Modification of Post Accident Sampling System Requirements", Prepared for C-E Owners Group by ABB Combustion Engineering, Inc., December 1991.

2) NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980, Items II.B.3 and II.F.1.

3) Letter from C. Y. Cheng (NRC) to W. T. Wagner (PASS Owners' Group), dated May 3, 1990.

4) Draft Commission Paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirement", Item I, Attachment to the Letter from D. M. Crutchfield (NRC) to E. E. Kintner ALWR Steering Committee), dated February 27, 1992.

## ABSTRACT

Several of the requirements of NUREG-0737 II.B.3, Clarification of TMI Action Plan Requirements, were reviewed. This report examines these requirements and how they are met by the Post Accident Sampling System (PASS). In some cases, alternative means of obtaining the required information are presented. Examples are: (1) The containment post-LOCA hydrogen monitors are proposed for analysis of containment hydrogen, (2) The need for heat tracing of the containment atmosphere sample line is obviated by the use of noble gases for Core Damage Assessment (CDA), and (3) RCS hydrogen/total gas is a secondary analysis with reliance primarily on control room instrumentation for the detection of bubble formation. For other topics, such as Sample Points and the Time Requirement, a clarification is presented. In all cases, the proposed methodology is consistent with the intent of the NUREG. This Topical Report was prepared by ABB Combustion Engineering Nuclear Power at the request of the Combustion Engineering Owners Group for review and approval by the Nuclear Regulatory Commission.

LIC-94-0169
Attachment

CEOG COMBUSTION ENGINEERING OWNERS GROUP

**CEN-415**
**Revision 1 - A**

# MODIFICATION
# OF
# POST ACCIDENT SAMPLING SYSTEM
# REQUIREMENTS

**Prepared for the**
**C-E OWNERS GROUP**
**SEPTEMBER, 1993**

ABB Combustion Engineering Nuclear Power

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## 1.0 INTRODUCTION

The purpose of this report is to identify alternative methods of meeting the intent of the requirements in NUREG-0737 II.B.3. The methodology described below represents a shift away from using the Post Accident Sampling System, PASS, as the primary means of obtaining chemical and radiochemical information in a post-accident environment. Instead, the proposed alternative methods will employ existing plant safety-grade equipment and take credit for analyses performed in the Final Safety Analysis Reports.

This topical report was prepared on behalf of the Combustion Engineering Owners Group. The report will form the basis of changes to be performed under 10CFR 50.59 which will be implemented by each utility.

## 2.0 BACKGROUND

A major concern during the accident at Three Mile Island in 1979 was the condition of the core. The extent of fuel melting and the size of the hydrogen bubble in the reactor vessel were two major unknowns. As a result, several regulations were promulgated which would ensure that (1) the extent of core damage could be determined and (2) appropriate operator action is taken in a timely manner to protect the public health and safety.

One of the principal products of this regulatory activity was NUREG-0737, Clarification of TMI Action Plan Requirements. A portion of this NUREG, section II.B.3, requires the ability to obtain chemical and radiochemical samples of the reactor coolant and containment atmosphere in a post-accident environment. This ability constitutes the Post Accident Sampling System, PASS. The purpose of PASS is to ensure that an assessment of the core

condition can be made in a timely manner. This provides important data so that the operators can take the appropriate action to minimize off-site radiation dose.

In addition to the requirement for PASS, several other regulations were implemented which would assist the operators in determining the extent of, and mitigating, core damage. Two notable hardware additions are the reactor vessel level monitoring system and the vessel head vent. These systems provide the operator with the capability to monitor the reactor vessel water level and a means of ensuring that the core remains covered during and after the acc[illegible]

Another Post-TMI hardware requirement was the installation of a post-LOCA containment hydrogen monitoring and recombiner system. The purpose of these redundant, safety-grade systems, is to measure and control the hydrogen concentration in the containment atmosphere to ensure that an explosive mixture is not formed.

The above serve as examples of existing capabilities which provide the operators with real-time information on the status and integrity of the core. As a result, several safety-grade systems perform many of the required functions in a more timely manner than PASS. The use of these systems forms the basis for meeting several of the requirements of NUREG-0737 II.B.3 while de-emphasizing the use of PASS.

## 3.0 METHODOLOGY

Each item to be considered for modification or relaxation is presented separately below. The purpose of the regulatory requirement is first reviewed. Where appropriate, an alternative means of obtaining the required information is then presented. Finally, the technical justification for the proposed change then is presented.

### 3.1 RCS pH

#### 3.1.1 Requirement

NUREG-0737 II.B.3 Criteria: There is no specific requirement to monitor RCS pH. Measurement of sump pH however, is recommended by Regulatory Guide 1.97.

#### 3.1.2 Purpose

The principal purpose of the pH measurement is to determine the corrosivity of the coolant in a post-accident situation. The objective is to maintain the pH of the containment sump water in the neutral to slightly alkaline region. This will minimize the propensity for cracking of austenitic stainless steels and minimize the corrosion of other structural materials in containment. In addition, maintaining the pH in the alkaline region will minimize the airborne radioiodine in containment.

Control of cracking of austenitic stainless steel is necessary to ensure the integrity of the plant safeguard systems. Minimization of corrosion of structural materials in containment, (principally aluminum and galvanized surfaces)

will minimize the quantity of hydrogen generated which could contribute to an explosive atmosphere.

3.1.3 Modification

Delete the measurement of sump pH. Rely on sump pH additive as analyzed in the FSAR. Retention of an archive sample for later analysis may be desirable. This sample could be the remnants of a sample taken for other analyses.

3.1.4 Justification

The pH of the containment sump water is controlled through the addition of sodium hydroxide or trisodium phosphate dodecahydrate (TSP). The delivery system is described in the FSAR. For those plants with a passive pH additive delivery system, (e.g. TSP baskets), the system was designed and analyzed to raise the sump solution to ≥ pH 7 without operator action. Limiting conditions for operation, and surveillance requirements exist the Technical Specifications in order to ensure the operability of this passive system. This typically consists of demonstrating the ability of a sample of TSP to raise the pH of a defined quantity of water from the Refueling Water Tank to ≥ pH 7 in less than 4 hours. This minimum pH value was selected in order to minimize the corrosive effects on both stainless steel and low alloy steels. In addition, maintaining the pH at ≥ 7 is required to minimize the volatility of iodine species.

In the case of stainless steels, most notably types 304 and 316, the principal concern is Stress Corrosion Cracking (SCC). Minimizing the potential for SCC is important in ensuring the integrity of the safeguard system piping.

Griess and Bacarella (Reference 5.1) performed a variety of experiments to evaluate the corrosion behavior of various materials in containment spray solutions. The tests were conducted to simulate a post-LOCA environment. A test of particular interest was run 9 with temperatures of 140, 100, and 55 °C for 24, 168, and 504 hours respectively. Run 9 was conducted in a chemical environment of 3000 ppm boron from boric acid and 6000 ppm sodium hydroxide. The resulting pH was 9.3 to 9.4. This solution was faulted with 100 ppm chloride (note that the technical specification for normal operation typically is $\leq$ 0.15 ppm). Specimens of mill annealed and sensitized types 304 and 316 stainless steel were subjected to the solution for the stated time. Griess and Bacarella reported that complete visual and selected metallographic examinations revealed no evidence of stress corrosion cracking.

The general corrosion of steel is of interest for the integrity of the many structural components fabricated from low alloy steel. Figure 1 from Reference 5.2 describes the general corrosion of steel as a function of pH in relative terms. Most notable is the small change in corrosion from pH 7 down to pH 4. This illustrates that slight deviations below pH 7 will have a negligible impact on the corrosivity of the sump solution.

The safety-grade ECCS in conjunction with the pH additive required by Technical Specifications minimizes the corrosivity of the sump solution. Therefore, the pH measurement via PASS is not necessary.

## 3.2 Containment Hydrogen

### 3.2.1 Requirement

NUREG-0737 II.B.3 Criteria: 2b, 8.

Criterion 2b requires the licensee to have the capability to quantify 'hydrogen levels in the containment atmosphere;'. Criterion 8 states in part, 'If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples,...'.

### 3.2.2 Purpose

Containment Hydrogen analytical results are used for the assessment of core damage and the potential formation of an explosive atmosphere in containment. The hydrogen in the containment atmosphere comes from a variety of sources including oxidation of zircaloy components, CVCS additive hydrogen, hydrogen from radiolytic decomposition of water, and hydrogen from corrosion of aluminum and zinc components in containment. The core damage assessment procedure should provide a means for attributing the hydrogen present to the various sources.

### 3.2.3 Modification

Use the current safety-grade containment hydrogen monitors for analysis and trending of the hydrogen concentration. Delete the requirement of the plant PASS system to measure containment hydrogen.

3.2.4 Justification

Containment hydrogen is best determined through the use of the current in-line safety-grade hydrogen monitors. This system provides redundant trains qualified for operation in a post-accident environment.

The benefits of using the data from these monitors are many. The in-line monitors provide real time data which can assist the operators in assessing core damage long before a grab sample could be obtained and analyzed. The safety-grade qualification ensures the accuracy of the results and minimizes the human factors associated with grab sample collection and analysis. Continuous sampling and trending of the results could be accomplished with no impact on worker dose. This method meets criterion 2b.

NUREG-0737 II.B.3 criterion 8 requires that the capability exist to obtain backup grab samples for all in-line monitored parameters. The intent of this requirement is to provide a defense-in-depth assurance that the desired sample and analysis can be obtained. This assurance is provided by the redundant safety-grade monitors which comprise the containment hydrogen monitoring system. Therefore, the intent of criterion 8 of NUREG-0737 II.B.3 is met.

## 3.3 Heat Tracing

3.3.1 Requirement

NUREG-0737 II.B.3 Criterion 11a.
Criterion 11 states in part that '...consideration should be given to the following items: (a) Provisions for purging

sample lines, for reducing plateout in sample lines,...'.

3.3.2 Purpose

The addition of heat tracing to the containment atmosphere sample line is in direct response to criterion 11a of NUREG-0737 II.B.3. Heat tracing minimizes the plateout of radioiodine species in the sample line.

3.3.3 Modification

Modify the plant Core Damage Assessment (CDA) procedure to perform CDA based upon noble gases. Delete the requirement for heat tracing of the containment atmosphere sample line.

3.3.4 Justification

The purpose of heat tracing for the containment atmosphere sample is to minimize the plateout of radioiodine species on the sample line walls. While radioiodine analysis results may be used for CDA, other radionuclides can provide the necessary information. The plant CDA procedure may utilize noble gases from the containment atmosphere sample for core damage assessment. The use of noble gases for core damage assessment is described in Reference 5.3. Noble gases are not susceptible to plateout and therefore a representative sample may be obtained without the use of heat tracing. In addition, this characteristic of noble gases will improve the accuracy of the sample as the need for a plateout correction is eliminated. Thus, the intent of Criterion 11a is met.

In addition to the above justification, the following NRC response to questions posed by the PASS Owners Group is

presented. These questions and responses were extracted from Reference 5.4.

'8. What are the NRC requirements for heat trace temperature on containment air sampling lines? What's too high and what's too low?

Response:

NRC does not have a requirement on heat trace temperature. The licensees are responsible to design and operate a sampling system that meets performance requirements identified in NUREG-0737 Item II.B.3. The temperature should be high enough to preclude condensation in the sample line. This temperature should be based on a review of the environmental conditions under which the sampling system must operate. Licensees should be aware that high heat trace temperature, in conjunction with the presence of chlorides, could result in sample line cracking as occurred at the Pilgrim Station.

9. Does the NRC still require pulling a post-accident iodine sample? What are the current thoughts about iodine sampling?

Response:

The NRC never required postaccident iodine sampling. A licensee that utilizes iodine in its core-damage assessment procedures must include appropriate design considerations to ensure representative sampling. A licensee that incorporated core-damage assessment procedures on the basis of representative containment atmosphere xenon and/or krypton activities need not

sample for iodine to meet NUREG-0737, Item II.B.3, PASS requirements.'

3.4 <u>RCS Hydrogen/Total Gas</u> **(See Note 1)**

3.4.1 Requirement

NUREG-0737 II.B.3 Criteria: 2c, 4.

Criterion 2 states that 'The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3 hour time frame established above, quantification of the following: ... (c) dissolved gases (e.g. $H_2$)...'. In addition, Criterion 4 states that 'Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or $H_2$ gas in the reactor coolant samples is considered adequate. Measuring the $O_2$ concentration is recommended but not mandatory.'

3.4.2 Purpose

The purpose of the dissolved gas/hydrogen measurement is two-fold: (1) provide an indication of potential bubble formation in the RCS, and (2) provide data for a core damage assessment.

This requirement was a direct result of the accident at TMI. Knowledge of the quantity of dissolved gases was considered important for the purpose of keeping the core covered with

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1) **This Section approved in part only. See SER for discussion.**

coolant. Depressurization could lead to the formation of a gas bubble. In the case of TMI however, there was no provision for monitoring the water level in the reactor vessel. In addition, there was no means of removing a bubble in TMI.

The principal source of dissolved hydrogen in a severe accident is that formed as a product of the zirc-water reaction. The hydrogen concentration, therefore, can be a direct indicator of the amount of fuel cladding failure.

3.4.3 Modification

Rely principally upon alternative means (existing systems and procedures) for monitoring bubble formation and core damage assessment. Retain hydrogen/total gas measurement as a backup method not subject to the 3-hour time requirement.

Employ the Reactor Vessel Level Monitoring System (RVLMS) and reactor head vent for monitoring and control of reactor vessel gas and coolant inventory. Rely on Emergency Procedures guidance for maintenance of natural circulation (Note: The Emergency Procedures should provide for the recovery from a loss of natural circulation in the event of bubble formation.). Perform core damage assessment using coolant radionuclide concentrations.

Modify the approach to RCS hydrogen/total gas sampling to include the option of using either the PASS or normal sampling system.

3.4.4 Justification

The proposed modification relies principally upon the use of the RVLMS and the radionuclide concentrations in the coolant. These will provide indications of a gas bubble in the RCS and information for a CDA respectively.

The measurement of RCS hydrogen concentration is useful only for accidents where the system remains pressurized. During a large break LOCA, when the RCS depressurizes, the gas concentration would become irrelevant from the perspective of both core uncovery and core damage assessment due to communication with the containment. In this scenario, the containment hydrogen monitors would provide information which could be used for the CDA.

During an accident where system pressure is maintained, one of the two major concerns is removal of heat from the core. Core heat removal is accomplished through the steam generators and/or the Emergency Core Cooling System (ECCS), via the Shutdown Cooling System. The formation of a gas bubble in the reactor vessel is a concern due to the potential for the bubble to disrupt flow in the core. The Reactor Vessel Level Monitoring System and reactor head vent system were developed specifically to monitor and correct this condition. The RVLMS gives the operator real-time data allowing an appropriate response to ensure adequate core cooling. This response may include the use of the head vent syscem to remove the gas bubble.

During a pressurized accident where forced circulation of the reactor coolant is lost, maintaining natural circulation for cooldown is of importance. Formation of a bubble in the steam

generators or reactor vessel may interrupt the natural circulation process. Combustion Engineering developed Emergency Planning Guides (EPGs) to assist the licensees in developing plant-specific procedures to respond to various accident scenarios. Guidance is provided in the EPGs (Reference 5.5 provides an example) to recover from loss of natural circulation. It is noteworthy that this guidance relies only on information available to the operator from control room indications. There is nc reliance upon PASS sample analytical results for operator action.

Based upon the justification above, it is recommended that the sampling and analysis for RCS hydrogen/total gas be retained, but de-emphasized. For those plants using an inline hydrogen monitor, RCS gas results are easily obtainable. However, a grab sample (including the backup grab sample required for the inline monitor) presents several difficulties which are discussed below with recommendations for an improved approach. The grab sample for RCS hydrogen/total gas should be considered of secondary importance relative to the RCS boron analysis and therefore should not be subject to the 3-hour time recommendation for the first sample.

One of the difficulties in obtaining meaningful grab samples of reactor coolant dissolve gas is the size of the sample bomb. In general, small volume sample bombs (typically 25 to 125 ml) are used to capture a pressurized reactor coolant sample. The small volume is required due to the high specific activity of the coolant in a design basis accident. The small volume allows a sample to be obtained while meeting the dose requirements of GDC 19. This sample size is sufficient for severe accidents where a large quantity of dissolved gas is present. However, for less severe accidents where core

damage, and consequently dissolved hydrogen, may be minimal, determining the hydrogen concentration from such a small volume of reactor coolant may be difficult (recall that the normal sampling system typically uses a 1 liter pressurized sample.

The following modification to the current exclusive use of the PASS for gas grab sampling is offered. Incorporate the option of using the PASS grab sampler or the normal RCS sampling system for gas determination. The PASS is used in the case of severe accidents where sufficient quantities of gas are present to provide a representative sample with confidence. The normal sampling system is used for cases where it is expected that lesser quantities of gas are present. The determination of which sample system is used is based upon the initial assessment of the extent of core damage either from gross area radiation monitoring or from the activity level of the first sample. Emergency response procedures could be written which correlate this activity level with an expected hydrogen concentration. Additionally, the activity level will allow a dose assessment to be performed for using the normal sampling system.

The use of the existing instrumentation previously noted, in conjunction with the improved option for grab sampling forms the basis for meeting the intent of Criterion 2c of NUREG-0737 II.B.3. Sufficient information is immediately available to the operator via control room indications for the RVLMS to ensure core cooling. Follow-up hydrogen analysis will provide additional information useful for plant recovery.

## 3.5 RCS Oxygen Analysis

### 3.5.1 Requirement

NUREG-0737 II.B.3 Criteria: 2.c, 4.

Criterion 2 states in part that onsite radiological and chemical analysis capability be established for quantification of '(c) dissolved gases (e.g. $H_2$), chloride...'. Criterion 4 states in part, 'Measuring the $O_2$ concentration is recommended, but is not mandatory.'

### 3.5.2 Purpose

The concentration of oxygen in the reactor coolant is of interest due to its influence in the cracking of austenitic stainless steels. Note that the analysis of the oxygen concentration is recommended by NUREG-0737 II.B.3, though not specifically required.

### 3.5.3 Modification

Delete the RCS oxygen analysis requirement from PASS. Obtain the RCS oxygen concentration via calculation from the RWT concentration or containment atmosphere concentration.

### 3.5.4 Justification

During a large break LOCA containment spray will be activated. The containment spray pumps will take suction from the containment sump and recirculate the fluid through the spray headers into the containment atmosphere. Coincident with this, the Low Pressure Safety Injection (LPSI) pumps also will

take suction from the sump and recirculate it through the core and out through the break. Through the common suction point, this will result in reactor coolant being circulated through the containment atmosphere.

In the above case, the concentration of oxygen in the reactor coolant can be calculated through the use of Henry's law. Knowing the containment temperature and pressure from the control room indications, and assuming an air-steam-hydrogen atmosphere at a saturated condition, the oxygen concentration is easily determined. It should be noted that knowledge of the RCS $O_2$ concentration during a large break LOCA is of limited value. Due to the communication between the RCS and containment atmosphere, no corrective action can be taken to reduce the oxygen concentration.

During a small break LOCA, the borated water source is the Refueling Water Tank (RWT). The oxygen concentration of the RWT may be determined directly by analysis. In the case of a tank open to the atmosphere, the RWT water may be assumed to be saturated and Henry's law may be used to determine the oxygen concentration in the reactor coolant.

## 3.6 Sample Points

### 3.6.1 Requirement

NUREG-0737 II.B.3 Criteria: 1, 11a.

Criterion 1 requires the licensee to '... have the capability to promptly obtain reactor coolant samples and containment atmosphere samples.'. Criterion 11a states in part, 'The postaccident reactor coolant and containment atmosphere

samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident.'.

3.6.2 Purpose

The purpose of the sample points is to ensure that adequate information regarding the condition of the core and the extent of the accident is available to plant personnel.

3.6.3 Modification

The sample points for PASS are limited to the containment atmosphere and the reactor coolant system.

3.6.4 Justification

The primary use of PASS is to evaluate the condition of the core. The information obtained from the reactor coolant and containment atmosphere samples can be used in the core damage assessment procedure. The results can be used to determine the extent of fuel melt and thereby provide information which can aid in the recovery of the plant. The indicated sample points are the only ones required to perform the stated function.

3.7 Time Requirement (**See Note 2**)

3.7.1 Requirement

NUREG-0737 II.B.3 Criterion: 1.

Criterion 1 states that, 'The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.'.

3.7.2 Purpose

The intent of the time requirement is to enable the plant personnel to determine the status of the core in a timely manner.

3.7.3 Modification

Clarify that the time requirement is, in fact, a recommendation which applies only to the first sample. Additionally, preparations such as suiting up, radiation measurements etc. may be performed prior to deciding to sample, and therefore, prior to 'starting the clock'.

3.7.4 Justification

A clarification was issued by the NRC in the form of a response to questions posed by the PASS Owners Group. The question and response relating to the time requirement were excerpted from reference 4.4 and are presented below.

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2) **This Section not approved. See SER for discussion.**

'6. When exactly does the 3-hour time limit start? If starting is based on the instruction to sample, what level of the utility passing that instruction "down" starts the clock? Have any utilities been fined due to not meeting the 3-hour limit?

In timing the three hours, what conditions must be assumed? Full response, dressing out, etc.?

Response:

For system design purposes, the 3-hour time limit starts at $t_o$ (core melt). This is to ensure that the capability exists to sample at maximum possible radiation conditions. In practice, the 3-hour time limit clock for PASS sampling, transport to laboratory and sample analyses starts when a decision is made to sample. The 3-hour time constraint applies to one sample (either reactor coolant or containment atmosphere), whichever is the more representative for the type of accident experienced. The second sample must also be taken, but not within the 3-hour time limit. The core damage assessment is also not included in the 3-hour time limit. Before the decision is made to sample, preparations may be made to get ready for PASS sampling (i.e., suiting up, radiation measurements, etc.).

The 3-hour time limit is a recommendation and not a requirement. If the sampling and analysis time is exceeded by about 15 minutes, the licensee should not be cited for a violation. No licensees have been issued a violation notice for an inability to meet the 3-hour limit.'

In addition to the above response by the NRC, the following justification is presented. The primary objective of plant operators after an accident is to stabilize the core. The basic tenets are reactivity control, heat removal, and maintenance of the integrity of the pressure boundary. In light of this, the only PASS sample parameter with any safety ramifications is the boron analysis to ensure adequate shutdown margin. However, evidence of shutdown is readily available to the operator via neutron monitors and temperature and pressure indications.

Emergency Procedure Guidelines (EPGs) were developed by Combustion Engineering for its owner utilities. These EPGs form the basis for the Emergency Operating Procedures (EOPs). The EPGs provide the utilities with the information necessary to mitigate the accident consequences. The EPGs perform this function with no reliance upon the Post Accident Sampling System. Operators rely upon plant instrumentation for information such as reactivity, heat removal, etc.

As a result, the sample analyses from PASS are confirmatory only. They do not influence operator action in the stabilization of the core. The analytical results may be useful for recovery operations.

## 4.0 CONCLUSION

Several of the requirements of NUREG-0737 Item II.B.3 were addressed. Based upon the justifications provided in the previous sections, it is concluded that the modifications meet the intent of stated sections of the NUREG. Adoption of these changes will provide the licensees with the guidelines necessary for improving the timely response to an accident.

In addition, the accuracy of some parameters (e.g. RCS hydrogen and core damage assessment via containment atmosphere) are improved. A further benefit is a reduction in worker dose by increased reliance on plant instrumentation.

## 5.0 REFERENCES

5.1 Griess, J. C., and Bacarella, A. L., The Corrosion of Materials in Reactor Containment Spray Solutions, Nuclear Technology, Vol. 10, April, 1971, pp. 546-553.

5.2 Uhlig, H. H., The Corrosion Handbook, John Wiley and Sons, New York, 1948, page 525.

5.3 CE-NPSD-241, Development of the Comprehensive Procedure Guideline for Core Damage Assessment, Task 467, C-E Power Systems, Combustion Engineering Inc., July 1983.

5.4 Letter, C. Y. Cheng, Chief, Materials and Chemical Engineering Branch, Nuclear Regulatory Commission, to William T. Wagner, Coordinator PASS Owners Group, dated 01/26/90.

5.5 Combustion Engineering Emergency Procedure Guidelines, Loss of Coolant Accident Recovery, CEN-152, Revision 03.

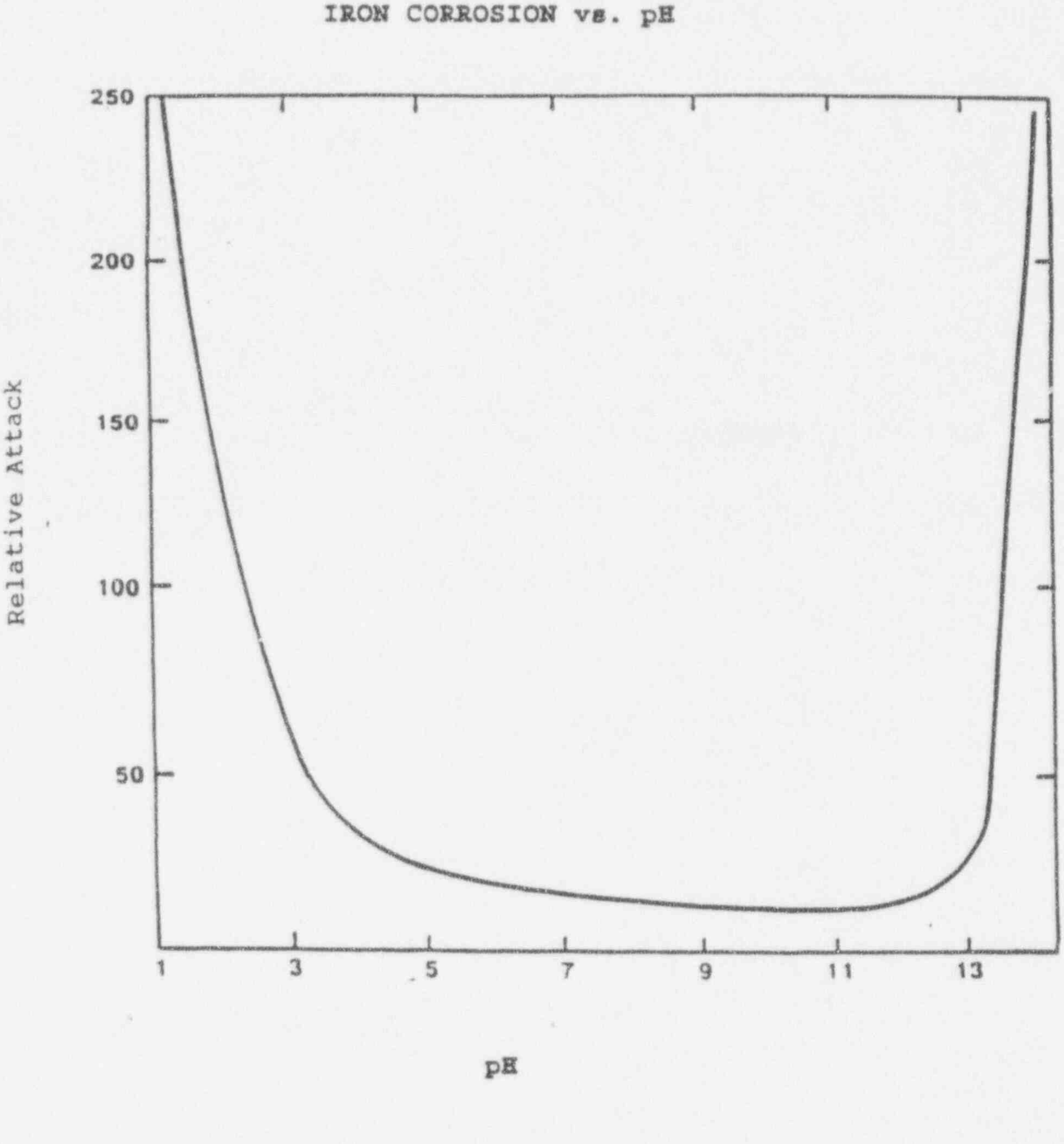


Figure 1

From Uhlig (Ref. 5.2) Corrosion of Iron by Water at 310°C (590°F) at Various Values of pH @ 25°C (Partridge and Hall, based upon data of Berl and van Taack)