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Special Report
September 1987

Guidelines for Permanent BWR Hydrogen Water Chemistry Installations—1987 Revision

Prepared by
Electric Power Research Institute
Palo Alto, California

R E P O R T   S U M M A R Y

SUBJECT	Nuclear plant corrosion control
TOPICS	Hydrogen water chemistry; BWR; Stainless steels; Stress corrosion; Water chemistry; Hydrogen
AUDIENCE	Generation engineers and managers

Guidelines for Permanent BWR Hydrogen Water Chemistry Installations—1987 Revision

Hydrogen water chemistry can effectively slow the rate of intergranular stress corrosion cracking in BWR piping. These NRC-approved guidelines afford utilities a safe, practical basis for designing, constructing, and operating permanent hydrogen water chemistry installations at BWR plant sites.

BACKGROUND Intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel piping in BWRs results in costly plant outages. Hydrogen water chemistry (HWC) reduces IGSCC by using feedwater additions of hydrogen to decrease the oxidizing power of reactor water and reduce its aggressiveness toward plant materials. The HWC process uses substantial quantities of hydrogen; therefore, its application requires an evaluation of the safety aspects of hydrogen storage and use at BWR plant sites, as well as consideration of applicable NRC licensing regulations. Preliminary guidelines for permanent BWR hydrogen water chemistry installations, prepared in 1986, discussed these safety aspects. However, the document required further revision to address NRC questions and concerns and to establish the guidelines as an industry standard appropriate for general distribution.

OBJECTIVE To develop generic NRC-accepted guidelines for the design, construction, and operation of HWC installations at BWRs.

APPROACH Industry personnel with experience in key technical areas and representing seven utilities, three manufacturers, and EPRI assembled to revise the guidelines. The group identified the safety, licensing, and operational factors necessary for the use and storage of hydrogen at a BWR plant. Members then researched and evaluated relevant codes, standards, regulations, and industrial practices and held several working sessions in 1986 to prepare, review, and revise the draft. Members of the BWR Owners Group for IGSCC Research reviewed and approved the revisions and the committee submitted them to NRC. The committee addressed NRC's ensuing questions and concerns, revised the draft, and resubmitted it for final approval.

RESULTS Designed primarily for utility use, this document also provides NRC with a standard for reviewing HWC installations. Organized in the general format of an industry standard, it discusses available, on-site hydrogen and oxygen supply options (compressed gas, cryogenic liquid, and electrolytic generation) as well as delivery system design and controls. It offers specifications for design, operation, maintenance, surveillance, radiation protection, and testing to provide for safe system and plant operation. The guidelines also suggest conservative methods for evaluating the hazards of hydrogen and oxygen storage. NRC has accepted these guidelines for utility use in implementing permanent HWC installations.

EPRI PERSPECTIVE An effective remedy for IGSCC in BWR piping, hydrogen water chemistry could also benefit the service performance of other plant components exposed to reactor coolant and will likely reduce the frequency of in-service piping inspections required by NRC. This document provides all information necessary to plan and implement a safe, permanent HWC installation. Approval of these generic guidelines by NRC in July 1987 has added substantial value by simplifying NRC review of individual utility installations.

Related EPRI reports NP-3959M, NP-4592-SR, and NP-5080 discuss HWC for BWRs.

PROJECT EPRI Project Manager: Robin Jones
BWR Owners Group / Nuclear Power Division

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Special Report, September 1987

Prepared by

BWR Owners Group for IGSCC Research
Hydrogen Installation Subcommittee

BWR Owners Group

and

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NOTICE

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

JUL 13 1987

Mr. J. H. Neils
Chairman, Regulatory Advisory Committee
BWR Owner's Group II for Intergranular
Stress Corrosion Cracking Research
414 Nicollet Mall
Minneapolis, Minnesota 55401

Subject: Acceptance for Referencing of Licensing Topical Report Titled, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," 1987 Revision

Dear Mr. Neils:

We have completed our review of the topical report submitted by your letter dated December 5, 1986.

We find the report acceptable for referencing in licensee requests for approval of permanent hydrogen water chemistry installation to the extent and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report is referenced in licensee requests for approval of a permanent hydrogen water chemistry installation, except to ensure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, we request that EPRI publish accepted versions of this report within three months of receipt of this letter. The accepted versions should incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted version should include an -A (designating accepted) following the report identification number.

Hydrogen water chemistry in combination with high water quality has demonstrated that mitigation and possibly complete suppression of intergranular stress corrosion cracking can be achieved in 280°C water at an electrochemical potential of less than -230 mV (Standard Hydrogen Electrode). This can be achieved with a dissolved oxygen content below 20 ppb and the conductivity maintained below 0.3 μS/cm. Consequently, the staff is developing criteria to give credit for effective hydrogen water chemistry in reducing frequency of inservice inspection of recirculation piping.

There have been a number of recent potentially hazardous hydrogen concentrations and/or deflagrations resulting from hydrogen leaks (NRC Information Notice No. 87-20: Hydrogen Leak in Auxiliary Building). When this topical report is revised to incorporate our evaluation, it may also be useful in providing industry guidance for the design, operation, maintenance, surveillance, and

G. Neils

testing of hydrogen supply systems (1) for providing a cover gas in the PWR volume control tank and (2) for cooling the generator (in addition to hydrogen water chemistry). We recommend that you consider providing such guidance in the final version.

Sincerely,

J E Richardson

James E. Richardson, Assistant Director
for Engineering
Division of Engineering & Systems Technology
Office of Nuclear Reactor Regulation

SAFETY EVALUATION REPORT

GUIDELINES FOR PERMANENT BWR

HYDROGEN WATER CHEMISTRY INSTALLATIONS

JULY 1987

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1 INTRODUCTION

1.1 Scope

The BWR Owner's Groups initially submitted the draft "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations" to the Director, NRC Office of Nuclear Reactor Regulation (NRR) on October 12, 1985. This staff's initial review indicated that the storage and use of large quantities of liquid hydrogen on a plant site raises the concern of potentially new and different accidents from those previously considered and evaluated as part of the facility licensing process.

In a letter to G. H. Neils, Chairman, Regulatory Advisory Committee, BWR Owner's Group II for IGSCC Research, (Bernaro, February 7, 1986), the staff indicated that licensees must consider whether proposed modifications in storage and use of relatively large quantities of liquid hydrogen and/or oxygen would result in hazards involving any of the three criteria for "an unreviewed safety question" defined in 10 CFR 50.59(a)(2). In a response (Neils, June 12, 1986) the Owner's Group requested a formal NRC staff review of all hydrogen and oxygen storage options (i.e., liquid hydrogen, liquid oxygen, and gaseous hydrogen).

A revised version of the "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations" was submitted to NRC for review (Neils, January 27, 1986), and the staff requested additional information concerning review of this submittal (Hulman, May 8, 1986). Another revision of the Guidelines (hereafter referred to as the Guidelines) incorporating responses to the staff's request for additional information was submitted (Neils, December 5, 1986) and is the basis for the review.

The electrolytic option that generates hydrogen and oxygen at the rate used in the process is not considered a storage option. Therefore, this report addresses liquid and gaseous hydrogen and liquid oxygen storage options and the electrolytic option.

1.2 Background

1.2.1 Hydrogen Water Chemistry

For IGSCC to occur in austenitic stainless steel, three conditions must exist simultaneously: a high stress region, a sensitized microstructure, and an adverse environment. In a boiling water reactor (BWR), the environment can mitigate the potential for IGSCC if the oxygen dissolved in the reactor coolant and its ionic impurity content are controlled.

BWR reactor coolant is demineralized water, typically containing 100 to 200 parts per billion (ppb) dissolved oxygen from the radiolytic decomposition of water. The electrochemical potential for BWR reactor coolant is near zero on the standard hydrogen electrode (SHE) scale. Even at a low conductivity (low ionic impurity), sensitized austenitic stainless steels are susceptible to IGSCC in 550°F (operating temperature) water at corrosion potentials near zero SHE. To mitigate the potential for IGSCC, the dissolved oxygen in the recirculating water can be reduced to less than 20 ppb by the addition of hydrogen to the feedwater. Dissolved hydrogen in the reactor coolant suppresses in-core radiolytic oxygen formation. BWR hydrogen water chemistry requiring control of oxygen to less than 20 ppb and a conductivity of less than 0.3 μS/cm will reduce the electrochemical potential to about -250 mV (SHE) resulting in a minimization of IGSCC. The (EPRI) BWR Owner's Group developed "BWR Water Chemistry Guidelines" (EPRI NP-3589-SR-LD), which must be met to obtain the full benefits of hydrogen water chemistry. These water chemistry guidelines also should be used as a basis for developing a plant-specific water chemistry control program. Hydrogen water chemistry appears to provide a means of suppressing both the initiation of IGSCC and the growth of preexisting cracks in sensitized stainless steel components in BWRs during power operation.

The Guidelines provide guidance for design, construction, and operation of permanent hydrogen addition systems at BWRs. Hydrogen water chemistry also requires an oxygen addition system that injects oxygen into the off-gas system to ensure that all excess hydrogen in the off-gas stream is recombined. Oxygen also may be needed for injection into the condensate and feedwater system to regulate reactor feedwater-dissolved oxygen between 20 to 50 ppb during power operation to minimize corrosion of the carbon steel in the condensate and feedwater system components. The Guidelines also document pertinent information on cryogenic oxygen storage and injection systems.

1.2.2 Hydrogen Explosion and Fire Experiences

Technical references in the Guidelines list approximately 100 incidents between 1921 and 1977 that produced flammable/explosive gas cloud releases. The potential dangers of explosive clouds are listed in the General Accounting Office report "Liquified Energy Gases Study," dated July 31, 1978.

National Aeronautic Space Administration has published a report (NASA TMX-71565, August 1974) describing incidents that occurred when liquified hydrogen was used as rocket engine fuel. Hydrogen deflagrations and explosions have occurred at reactor sites when gas storage tanks were being filled. An internal hydrogen tanks detonation also occurred at Los Alamos when a stream of oxygen accidentally leaked from a high pressure source into the hydrogen storage cylinder (Investigation Report, June 3, 1981).

Experimental liquid hydrogen spill tests indicate that the cryogenic liquid release to the ground will create a dense heavier-than-air plume that can travel up to 1500 feet before absorbing heat and gaining buoyancy (Authur D. Little, Inc., March 22, 1960). This cloud has regions of both explosive and flammable concentrations. National Bureau of Standards Monograph 168 indicates hydrogen is flammable in air in the range of 4.0-75.0 vol % and detonable in air in the range of 18.3 - 59.0 vol %. One gallon of liquid hydrogen has the explosive energy equivalence of 1.37 lbs of TNT (1 lb hydrogen is equivalent to 2.4 lbs TNT) in an open air explosion. One thousand scf of gaseous hydrogen is equivalent to 27.1 lbs of TNT.

1.2.3 Regulatory Concerns

During the past two decades the Atomic Energy Commission (AEC) and NRC have evaluated man-made hazards in the vicinity of nuclear power plants. These potential hazards have included the transport and nearby storage of munitions, explosives, toxic gases, and explosive flammable gases. When such hazards have a sufficiently high probability of occurring, the plant's structures, systems, and components important to safety must be designed to withstand the possible effects of explosions or toxic gases without damage that would prevent a safe and orderly shutdown of the plant. Guidelines for the evaluation of these potential hazards are identified in the NRC's Standard Review Plan (SRP) (NUREG-0800) (Sections 2.2.1 and 2.2.2, "Identification of Potential Hazards in Site Vicinity," and 2.2.3, "Evaluation of Potential Accidents"). Regulatory Guide (RG) 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," Revision 1, describes vapor cloud explosions.

The Guidelines must address the potential impact of inadvertent releases or failures in hydrogen and oxygen storage and/ or injection systems on plant safety systems. The siting of hydrogen and oxygen storage facilities must be prescribed so that explosions and fires will not affect safety-related structures.

A second regulatory concern is the increased N-16 activity in the steam due to hydrogen injection. In normal BWR water chemistry, N-16 combines with oxygen to form water-soluble, nonvolatile nitrates and nitrites. However, when hydrogen is injected into the feedwater, N-16 forms a more volatile species (NH_3). Therefore, the steam phase N-16 levels are increased. Appropriate changes to the radiation protection program may be needed to compensate for increased radiation levels and to maintain exposures as low as is reasonably achievable (ALARA).

1.3 General System Description

The hydrogen water chemistry system is composed of a hydrogen supply, an oxygen supply, and hydrogen and oxygen injection systems. Hydrogen is supplied as a high pressure gas or as a cryogenic liquid. Oxygen is supplied as a cryogenic liquid. Hydrogen and oxygen can also be generated on site by the dissociation of water by electrolysis. (The electrolytic method is not a storage option and is, therefore, not reviewed in this report). Cryogenic hydrogen and oxygen are stored in vacuum-jacketed vessels. The liquified gases are vaporized by the use of ambient air vaporizers before the gases are pumped to the injection system. The gaseous hydrogen storage bank consists of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) gas storage vessels. The hydrogen and oxygen systems include flow control and flow measuring equipment and necessary instrumentation and controls to ensure safe, reliable operation. Hydrogen gas is injected into the suction of the feedwater or condensate booster pumps to provide adequate mixing and dissolution.

Gaseous oxygen is injected into the portion of the off-gas system that is already diluted so that the addition of oxygen does not create a combustible mixture. The injected oxygen ensures that all excess hydrogen in the off-gas stream is recombined.

1.4 Design Criteria

The hydrogen water chemistry system is not safety related. Equipment and components need not be redundant (except where required to meet good engineering practice), seismic Category I, electrical Class 1E, or environmentally qualified. However, proximity to safety-related equipment or other plant systems requires special consideration in the design, fabrication, installation, operation, and maintenance of hydrogen and oxygen addition systems. Hydrogen gas and cryogenic hydrogen and oxygen storage tanks are designed, fabricated, tested, and stamped in accordance with Section VIII, Division 1, of the ASME Code for unfired pressure vessels.

1.5 Hydrogen Storage Facilities

Gaseous hydrogen is stored in seamless ASMF Code vessels at pressures up to 2400 psig and ambient temperature. Transportable vessels, which can also be used for gaseous hydrogen storage at 2650 psig and ambient temperature, are designed to Department of Transportation standards. The Guidelines cover tank sizes from 1000 to 14,000 scf. With either type of storage, the gas is routed through a pressure control station that maintains a constant hydrogen supply pressure to the hydrogen injection system. The tube bank should be supported to prevent movement in the event of line failure, and each tube should be equipped with a close-coupled shutoff valve. As an alternative, one safety valve per bank of tubes can be used, provided the safety valve is sized to handle the maximum relief from all tubes tied into the valve. The pressure control station should be of a manifold design with two full-flow parallel pressure-reducing regulators. An excess-flow check valve should be installed in the manifold immediately downstream of the regulators to limit the flow rate in the event of a line break. A tube trailer grounding assembly should be provided at each discharge stanchion to ground the tube trailer before hydrogen is transferred.

Liquid hydrogen is stored in a vacuum-jacketed vessel with a capacity of up to 20,000 gallons at pressures up to 150 psig and temperatures up to -403°F (saturated). In addition to ASME Code inspection requirements, inner vessel longitudinal welds should be examined radiographically. For overpressure protection, dual full-flow safety valves and emergency backup rupture discs are provided. Hydrogen tanks and delivery vehicles should be grounded, and the storage system should be protected from the effects of lightning. Excess flow protection should be added wherever a line break would release a quantity of hydrogen large enough to threaten safety-related structures. The liquid hydrogen will be vaporized by the use of ambient air vaporizers.

1.6 Oxygen Storage Facilities

Liquid oxygen is stored in a vacuum-jacketed vessel at pressures up to 250 psig and temperatures up to -251°F (saturated) with capacities between 3,000 and 11,000 gallons. Oxygen removed from storage vessels should be vaporized through ambient air vaporizers and routed through a pressure control station that maintains gas pressures within the desired range of the injection system. Overpressure protection of the storage tank is provided by dual full-flow safety valves and emergency backup rupture discs.

1.7 Gas Injection Systems

Excess flow valves should be installed at appropriate locations in the hydrogen line to restrict flow out of a broken line. To meet this requirement other options are that hydrogen lines in safety-related areas should either be designed to seismic Class I requirements or sleeved so that the outer pipe is directly vented to the outside (Branch Technical Position CMEB 9.5-1, Revision 2, July 1981, SRP 9.5.1). Feedwater hydrogen injection lines should contain a check valve to prevent feedwater from entering the hydrogen line and to protect upstream hydrogen gas components. Automatic isolation valves should be installed in each injection line to prevent hydrogen injection into a non-operating feedwater pump. Purge connections should be provided to completely purge air from the system before hydrogen is released into the line. Area hydrogen monitors should be located at high points where hydrogen may collect and above components where potential hydrogen leaks may occur. Hydrogen monitors should be located so that they can detect hydrogen with or without normal ventilation. System design should conform with pertinent sections of 10 CFR 50.48.

1.8 Instrumentation and Control

The instrumentation should (a) provide indication and/or recording of parameters necessary to monitor and control the hydrogen injection system and (b) indicate and/or alarm abnormal or undesirable conditions. Parallel flow control valves should be provided in the hydrogen injection line for system reliability and maintainability. The recommended trips of the hydrogen and oxygen injection system include: reactor scram, low residual oxygen in the off-gas, high area hydrogen concentration, low oxygen injection system supply pressure or flow, off-gas train or recombiner train trip, and high hydrogen flow.

Provisions should be made to continuously monitor the dissolved oxygen in the reactor coolant. The off-gas flow downstream of the recombiners should be continuously monitored for hydrogen and oxygen.

2 EVALUATION

2.1 Site Characteristics for Gaseous and Liquid Hydrogen Storage

The Guidelines reference the National Fire Protection Association (NFPA) standards 50A and 50B for the location of gaseous and/or liquid hydrogen supply systems, respectively. These include

* Ready access to delivery equipment and to authorized personnel; suitable roadways or other means of access for emergency equipment, such as fire department apparatus, shall be provided.

* Storage containers shall not be located under electric power lines or where they would be exposed should the lines fail.

* Storage containers shall not be located close to piping containing flammable or combustible liquids, flammable gases, or piping containing oxidizing material.

* Where it is necessary to locate the hydrogen containers on ground that is level with or lower than adjacent flammable and combustible liquid storage or oxygen storage, suitable protective means shall be taken (such as diking, diversion curbs, or grading) to prevent accumulation of liquids within 50 feet of the storage container. Liquified hydrogen storage containers should be located on ground higher than flammable and combustible liquid storage or liquid oxygen storage.

Other considerations for siting include

* The route used for hydrogen delivery on site should be appropriate.

* The storage facility shall be completely fenced, even when located in a security area, and it should be lighted to facilitate night surveillance.

* Truck barriers shall be installed around the perimeter of the storage facility for protection in case of vehicular accidents.

* The hydrogen storage facility shall be located so there is adenuate separation between it and safety-related structures so that explosion and fire overpressures and thermal fluxes are within design considerations.

* Air pathways into safety-related structures should exceed a minimum separation distance so that the release from a possible pipe break is below the lower flammability limit of 4% before reaching the air pathway into safety-related structures.

2.2 Site characteristics for Liquid Oxygen Storage

The Guidelines reference NFPA 50 standards for the location of liquid oxygen storage systems. These include

* There shall be ready access to mobile supply equipment, at ground level, to authorized personnel.

* The location selected shall not be beneath electric power lines, piping containing all classes of flammable or combustible liquids, or piping containing flammable gases, nor should it be an area that would be impacted by the failure of these components.

* Noncombustible material surfacing shall be provided in an area extending at least 3 feet from points at ground level on which liquid oxygen might fall during operation of the system and filling of the storage container.

* When a liquid oxygen storage facility is on ground lower than all classes of adjacent flammable and liquid storage, suitable means shall be provided (such as by diking, diversion curbs, or grading) to prevent accumulation of flammable or combustible liquids under the oxygen storage facility.

Other considerations for siting include

* The route used for liquid oxygen delivery on the site should be appropriate.

* The storage facility shall be completely fenced, even when located within the security area, and it shall have lighting to facilitate night surveillance.

* Truck barriers shall be installed around the perimeter of the storage facility for protection in case of vehicular accidents.

* Liquid oxygen storage facility shall be located so that ingestion of oxygen-enriched atmospheres (above 30 volume %) into safety-related air intakes is not possible in the event of an oxygen spill.

2.3 Meteorological Considerations

A massive failure of a large pressurized cryogenic hydrogen storage tank would result in a turbulent release of the gas that may result in a fire or explosion.

To reduce the potential for impact on plant safety structures, the storage facility should be far enough from the safety structures so any overpressure it experiences from an explosion would not exceed that from hurricane or tornado winds.

Unconfined hydrogen-air mixtures generally burn rapidly, but without detonation, when they are initiated by heat, spark, or flame unless there is flame acceleration as a result of obstacles. In this case a deflagration/detonation transition may occur. Because hydrogen diffuses rapidly in air, it will not form persistent flammable mixtures when the gaseous hydrogen is released in open, unconfined areas. However, in confined areas, or when ignition of the hydrogen-air mixture is caused by a shock source equivalent to a blasting cap or small explosive charge, the mixture can detonate. Liquid hydrogen releases can produce dense plumes with flammable/detonable concentrations that can travel hundreds of feet before being diluted to a non-hazardous mixture.

An additional consideration is of the prevailing wind flow. A hydrogen leak in the presence of winds can lower the probability of a flammable or explosive environment near or at plant air intakes. The meteorological measurement program required at nuclear plants should serve as the source of this wind direction information.

Slow leaks of hydrogen gas outdoors or in unconfined areas tend to mix with the ambient air and not result in flammable or explosive mixtures.

The meteorological and siting considerations presented in the Guidelines are acceptable as a basis for establishing onsite hydrogen storage. Before individual plant facilities are installed the prevailing winds and structure locations should be reviewed.

2.4 Gaseous Hydrogen Safety Considerations

The Guidelines are based on the safety analysis of the failure of single vessels and do not address simultaneous failure of multiple storage vessels. In the case of the Los Alamos tube trailer, hydrogen explosion of a single tube did not damage the adjacent hydrogen vessels. This event provides a technical basis for assuming only single vessel failure (Investigation Report, June 3, 1981). At two reactor sites hydrogen explosions and fireballs during filling operations occurred over the storage tanks but did not damage the adjacent cylinders (Reportable Event No. 07953, March 5, 1987, NUREG/CR-3551, May 1985).

When a gaseous storage vessel ruptures, the expansion of the high-pressure gas results in turbulent mixing with the surrounding air. For hydrogen, the bulk of the release will go through the detonation limits before the wind can produce an explosive concentration plume that could travel far from the vicinity of the storage tank area.

The hydrogen storage area should be at a sufficient distance from safety-related structures so that the thermal flux from the burning hydrogen gas fire-ball or the blast overpressure from hydrogen detonation will not cause failure of the safety-related structures.

The staff has performed independent calculations and evaluations that confirm the following figures in the Guidelines for gaseous hydrogen storage systems:

* Figure 4-1, thermal flux vs. distance from fireball center
* Figure 4-2, minimum required separation distances to safety-related structures versus vessel size
* Figure 4-3, minimum required separation distance (to air pathways into safety-related structures) versus ID of pipe for release from 2450 psig gaseous hydrogen

The Guidelines recommend, in Appendix B, a method to determine separation distances for hydrogen storage to prevent damage to nuclear power plant safety structures in the event of a hydrogen explosion. Appendix B is based on earlier work performed by Sandia National Laboratories for NRC (NUREG/CR-2462). These recommendations are applicable for reinforced concrete or masonry walls that are at least 8 inches thick. Other structures with light-guage metal paneling walls and metal tanks should be evaluated on a case-by-case basis.

The staff reviewed the separation distance for hydrogen storage facilities with 8-inch or greater reinforced concrete or reinforced masonry walls (upper curve in Figure 4-2 and 4-5 of the Guidelines). This curve is based on the British Explosives Storage and Transport Committees recommendations (New York Academy of Sciences Annals, Vol. 1, 152, 1968). On the basis of this review, the staff concludes that the recommendations are reasonable and valid. The staff finds there are ample data from well-documented explosion experiments, damage records from accidental explosions, and war-time experiences (bomb damage), all of which were considered in the formulation of the recommendations.

The Guidelines provide separation distance from hydrogen storage facilities for 18-inch or greater reinforced concrete walls (curves (a) and (b) in Figures 4-2 and 4-5 of the Guidelines). Curves (a) and (b) are applicable for the indicated static pressure capacities and tensile steel factors, and are acceptable by the staff. The method of analysis for constructing these curves is conventional and generally follows the guidelines of the American Society of Civil Engineers (ASCE) Manual No. 58 (1980) and American Concrete Institute (ACI) 349-80 (April 1981).

The staff has not formally reviewed nor accepted the ASCE Manual. Special provisions for impulsive (blast) and impactive (missile) effects for concrete structures were addressed in the ACI 349-80 (April 1981) which has been accepted by the staff with the exception of certain ductility ratios. Appendix A to SRP 3.5.3) provides guidance for design of both steel and reinforced concrete structural elements (e.g., missile barriers, columns, slabs) subject to impactive or impulsive loads, such as impacts due to missiles or blasts. Ductility ratios for structural steel members are given in Appendix A to SRP 3.5.3. For reinforced concrete members, the requirement of ductility ratios is specified in RG 1.142, Revision 1. American Concrete Institute (ACI) 349-80 is to be used in conjunction with RG 1.142, Revision 1, for reinforced concrete structures, and American Institute of Steel Construction (AISC) Specification ("Manual of Steel Construction") is to be used in conjunction with Appendix A to SRP 3.5.3 for steel structures. Because curves (a) and (b) in Figures 4-2 and 4-5 in the Guidelines comply with RG 1.142, they are acceptable if the ductility ratio is limited to 3.

The staff concludes that curves (a) and (b) in the Guidelines can be used for determining the separation distance for reinforced concrete walls from gaseous and liquid hydrogen storage facilities. Walls with different static pressure capacities and/or tensile steel factors can use the methods in Appendix B to the Guidelines, pages 10 through 13.

2.5 <u>Liquid Hydrogen Safety Considerations</u>

The major hazard from the storage and use of large quantities of cryogenic liquid hydrogen on reactor sites is that of producing flammable/explosive clouds that can drift near or be taken into air ventilation systems of safety-related structures. Cryogenic hydrogen released to the environment will form a dense heavier-than-air plume that will drift along with wind currents and by gravity to lower elevations until it gains sufficient heat to produce buoyancy. Experimental data indicate plume travel of the order of 1000 feet from a liquid hydrogen flow rate of 0-18 Kg/sec.

The staff has performed independent calculations to check the values shown in the Guidelines. The staff used NASA data to check the thermal flux data (Figure 4-4 in the Guidelines). Standard meteorological data were used to check the hydrogen concentrations at the nearest safety-related air intakes (Figure 4-6 in the Guidelines). The staff used the Guidelines (Hoehne and Luck, 1970) to determine the lower flammable concentrations from various sizes of pipe breaks in gaseous hydrogen lines. In addition, the staff noted blast overpressure effects on both reinforced brick houses and reinforced concrete houses from nuclear weapons tests. The staff also observed that the 5 psi overpressure that destroyed an unreenforced brick house had no effect on a reinforced concrete house that had been designed to comply with California Code for earthquake-resistant construction (Glasstone, 1962). These data indicate that Figure 4-5 of the Guidelines is conservative when it is applied to safety-related structural walls.

Licensees may use the minimum separation distance curves in Figure 4-5 of the Guidelines in requests for approval of permanent hydrogen water chemistry installations.

2.6 Liquid Oxygen Safety Considerations

The major threat from the release of cryogenic liquid oxygen is the formation of dense plumes that disperse by slumping (due to gravity) and by motion of existing winds. The potential for oxygen clouds reaching flammable materials or entering safety-related air intakes should be avoided. Oxygen will not explode and is nonflammable, but ignition of combustible materials may occur more readily in an oxygen-rich atmosphere than in air.

The liquid oxygen tank capacity versus distance curves were checked by independent staff analysis and found to be acceptable (Figure 4-8 of the Guidelines). The recommended separation distances between liquid storage tanks and safety-related air intakes are reasonable. The separation distances are such that the vapor cloud released from a failed tank would disperse sufficiently so that the oxygen content at the air intakes will not support increased combustibility of ignitible materials.

2.7 Radiation Protection/ALARA Program

The staff has also reviewed the Guidelines to ensure that the dose rate increase in plant areas due to N-16 equilibrium changes during hydrogen addition has been considered in plant operation procedures. To reduce workers' doses, the Guidelines uses a programmatic approach that outlines additional health physics procedures and that is intended to augment current plant radiation protection procedures (current procedures would not change). Specifically, the Guidelines walls that are at least 8 inches thick. Other structures with light-guage metal paneling walls and metal tanks should be evaluated on a case-by-case basis.

The staff reviewed the separation distance for hydrogen storage facilities with 8-inch or greater reinforced concrete or reinforced masonry walls (upper curve in Figure 4-2 and 4-5 of the Guidelines). This curve is based on the British Explosives Storage and Transport Committees recommendations (New York Academy of Sciences Annals, Vol. 1, 152, 1968). On the basis of this review, the staff concludes that the recommendations are reasonable and valid. The staff finds there are ample data from well-documented explosion experiments, damage records from accidental explosions, and war-time experiences (bomb damage), all of which were considered in the formulation of the recommendations.

The Guidelines provide separation distance from hydrogen storage facilities for 18-inch or greater reinforced concrete walls (curves (a) and (b) in Figures 4-2 and 4-5 of the Guidelines). Curves (a) and (b) are applicable for the indicated static pressure capacities and tensile steel factors, and are acceptable by the staff. The method of analysis for constructing these curves is conventional and generally follows the guidelines of the American Society of Civil Engineers (ASCE) Manual No. 58 (1980) and American Concrete Institute (ACI) 349-80 (April 1981).

The staff has not formally reviewed nor accepted the ASCE Manual. Special provisions for impulsive (blast) and impactive (missile) effects for concrete structures were addressed in the ACI 349-80 (April 1981) which has been accepted by the staff with the exception of certain ductility ratios. Appendix A to SRP 3.5.3) provides guidance for design of both steel and reinforced concrete structural elements (e.g., missile barriers, columns, slabs) subject to impactive or impulsive loads, such as impacts due to missiles or blasts. Ductility ratios for structural steel members are given in Appendix A to SRP 3.5.3. For reinforced concrete members, the requirement of ductility ratios is specified in RG 1.142, Revision 1. American Concrete Institute (ACI) 349-80 is to be used in conjunction with RG 1.142, Revision 1, for reinforced concrete structures, and American Institute of Steel Construction (AISC) Specification ("Manual of Steel Construction") is to be used in conjunction with Appendix A to SRP 3.5.3 for steel structures. Because curves (a) and (b) in Figures 4-2 and 4-5 in the Guidelines comply with RG 1.142, they are acceptable if the ductility ratio is limited to 3.

The staff concludes that curves (a) and (b) in the Guidelines can be used for determining the separation distance for reinforced concrete walls from gaseous and liquid hydrogen storage facilities. Walls with different static pressure capacities and/or tensile steel factors can use the methods in Appendix B to the Guidelines, pages 10 through 13.

2.5 Liquid Hydrogen Safety Considerations

The major hazard from the storage and use of large quantities of cryogenic liquid hydrogen on reactor sites is that of producing flammable/explosive clouds that can drift near or be taken into air ventilation systems of safety-related structures. Cryogenic hydrogen released to the environment will form a dense heavier-than-air plume that will drift along with wind currents and by gravity to lower elevations until it gains sufficient heat to produce buoyancy. Experimental data indicate plume travel of the order of 1000 feet from a liquid hydrogen flow rate of 2-18 kg/sec.

The staff has performed independent calculations to check the values shown in the Guidelines. The staff used NASA data to check the thermal flux data (Figure 4-4 in the Guidelines). Standard meteorological data were used to check the hydrogen concentrations at the nearest safety-related air intakes (Figure 4-6 in the Guidelines). The staff used the Guidelines (Hoehne and Luck, 1970) to determine the lower flammable concentrations from various sizes of pipe breaks in gaseous hydrogen lines. In addition, the staff noted blast overpressure effects on both reinforced brick houses and reinforced concrete houses from nuclear weapons tests. The staff also observed that the 5 psi overpressure that destroyed an unreonforced brick house had no effect on a reinforced concrete house that had been designed to comply with California Code for earthquake-resistant construction (Glasstone, 1962). These data indicate that Figure 4-5 of the Guidelines is conservative when it is applied to safety-related structural walls.

Licensees may use the minimum separation distance curves in Figure 4-5 of the Guidelines in requests for approval of permanent hydrogen water chemistry installations.

2.6 Liquid Oxygen Safety Considerations

The major threat from the release of cryogenic liquid oxygen is the formation of dense plumes that disperse by slumping (due to gravity) and by motion of existing winds. The potential for oxygen clouds reaching flammable materials or entering safety-related air intakes should be avoided. Oxygen will not explode and is nonflammable, but ignition of combustible materials may occur more readily in an oxygen-rich atmosphere than in air.

The liquid oxygen tank capacity versus distance curves were checked by independent staff analysis and found to be acceptable (Figure 4-8 of the Guidelines). The recommended separation distances between liquid storage tanks and safety-related air intakes are reasonable. The separation distances are such that the vapor cloud released from a failed tank would disperse sufficiently so that the oxygen content at the air intakes will not support increased combustibility of ignitible materials.

2.7 Radiation Protection/ALARA Program

The staff has also reviewed the Guidelines to ensure that the dose rate increase in plant areas due to N-16 equilibrium changes during hydrogen addition has been considered in plant operation procedures. To reduce workers' doses, the Guidelines uses a programmatic approach that outlines additional health physics procedures and that is intended to augment current plant radiation protection procedures (current procedures would not change). Specifically, the Guidelines recommend an appropriate ALARA commitment for plant management, an initial and continuous radiation survey program, potential plant shielding changes, and potential maintenance activities. These programmatic procedures, in addition to normal plant radiation protection procedures, are sufficient to ensure that during hydrogen addition the plant will continue to meet the requirement of 10 CFR 20 and the recommendations of RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be as Low as is Reasonably Achievable." Thus the procedures are acceptable. These procedures will also ensure compliance with site boundary radiological limits required by 40 CFR 190.

2.8 Main Steam Line Radiation Monitoring

The staff reviewed the impact of the proposed changes on previously approved safety analyses of anticipated operational occurrences and postulated accidents.

The main steam line radiation monitors (MSLRMs) provide reactor scram and reactor vessel and primary containment isolation signals when high-activity levels are detected in the main steam lines. Additionally, these monitors serve to limit radioactivity release in the event of fuel failures. Technical Specification (TS) changes are needed to accommodate the expected increase in main steam activity levels (from increased N-16 levels in the steam phase) as a result of hydrogen injection into the primary system.

The BWR Owners Group state that the only transient or postulated accident that takes credit for the main steam line high radiation scram and isolation signals is the control rod drop accident (CRDA). The staff notes that for a CRDA, the primary function of the MSLRMs is to limit the transport of activity released from failed fuel to the turbine and condensers by initiating closure of the main steam isolation valves and thus isolating the reactor vessel. Main steam line high radiation will also produce a reactor scram signal and will isolate the mechanical vacuum pump and the gland seal steam exhaust system to reduce leakage of fission products to the atmosphere from the turbine and condensers. Reactor scram in the event of a CRDA, however, would be initiated by signals from the neutron monitoring system.

Generic analyses of the consequences of a CRDA have shown that fuel failures are not expected to result from a CRDA occurring at greater than 10% power (Stirn et al., March 1972; Strin et al., January 1973; Strin et al., July 1972). This is primarily a result of analyses that show that as power increases, the severity of the CRDA rapidly decreases as a result of the effects of increased void formation and increased Doppler reactivity feedback. The hydrogen injection will be restricted to power levels above 20% of rated power for all plants.

Main steam line radiation levels can increase up to approximately fivefold with hydrogen water chemistry. The majority of BWRs have a TS requirement for the MSLRM setpoint that is less than or equal to three times the normal rated full-power background. For these plants, an adjustment in the MSLRM setpoint may be required to allow operation with hydrogen injection. For earlier BWRs with MSLRM setpoints of 7 to ten 10 times normal full-power background, a setpoint change may not be required with hydrogen water chemistry.

For plants at which credit is taken for an MSLRM-initiated isolation in the CRDA, a dual setpoint approach may be used. At most plants, the MSLRM setpoint is specified in the plant TS as some factor times rated full-power radiation background. With hydrogen addition, the full-power background could increase up to five times that without hydrogen addition. Below 20% of rated power or the power level required by the FSAR or TS, the existing setpoint is maintained at the TS factor above normal full-power background, and hydrogen should not be injected. At about 20% of rated power, the MSLRM setpoint should be readjusted to the same TS factor above the rated full-power background with hydrogen addition. This adjustment will be made by the plant personnel during startup and shutdown. Plant power will remain constant during this adjustment process. Thus, the TS factor by which the MSLRM setpoint is adjusted remains the same with and without hydrogen addition, but the background radiation level increases with hydrogen addition. If an unanticipated power reduction event occurs so that the reactor power is below 20% without the required setpoint change, control rod motion should be suspended (except for scram or other emergency actions) until the necessary setpoint adjustment is made. TS changes will be required to suspend control rod motion during setpoint adjustment.

On the basis the discussion above, the staff finds that section 8 of the Guidelines is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and this technical evaluation.

3 CONCLUSION

On the basis of the above evaluation, the staff finds that the Licensing Topical Report, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," 1987 revision, is acceptable for reference in future licensee requests for approval of permanent hydrogen water chemistry installations. The basis for this acceptance is that the Guidelines meet the applicable requirements and guidance from the following regulatory guides, standard review plan sections, branch technical positions, and federal regulations:

- Regulatory Guide 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," Revision 1, February 1978

- Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," Revision 1, October 1981

- Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be as Low as is Reasonably Achievable," Revision 3, June 1978

- Standard Review Plant Section 9.5.1, Branch Technical Position CMEB 9.5-1 "Guidelines for Fire Protection for Nuclear Power Plants," July 1981

- Standard Review Plan Section 3.5.3, "Barrier Design Procedures," Revision 1, July 1981

- 10 CFR 50.48, "Fire Protection"

- 40 CFR 190, "Protection Environment, Environmental Radiation Protection Standards for Nuclear Power Operations"

A licensee request for approval for a permanent hydrogen water chemistry installation that incorporates this Licensing Topical Report by reference should include the following information:

- Any exceptions or deviations from the "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," 1987 Revision, Licensing Topical Report

- Justification that any exceptions or deviations from the Guidelines will not affect the safety of the plant or the public

- The maximum quantity of stored gaseous hydrogen and/or liquid hydrogen and oxygen and its distance from safety-related structures

- Technical Specification changes, if required, to accommodate the expected increase in main steam line radiation setpoint

- A description of hydrogen and oxygen storage facilities, including safety features

- A description of hydrogen and oxygen injection subsystems, including instrumentation, controls, and safety features

- The delivery route of hydrogen and oxygen supply tank trucks on site, including truck tank capacity.

- A radiological protection program to ensure that radiological exposures to plant personnel and the general public are consistent with ALARA requirements

- A discussion on implementation of BWR Owner's Group Water Chemistry Guidelines

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ABSTRACT

Intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel piping in BWRs has resulted in costly plant outages. One method shown effective in arresting pipe cracking and pipe crack growth is a process known as Hydrogen Water Chemistry (HWC). HWC consists of maintaining good water chemistry and adding hydrogen to the feedwater. Addition of hydrogen decreases the oxidizing power of the reactor water and reduces its aggressiveness toward plant structural materials. This document provides guidelines for design, construction, and operation of permanent hydrogen injection systems at BWRs. The scope of this document includes the currently available on-site hydrogen and oxygen supply options (i.e., compressed gas, cryogenic liquid, and electrolytic generation) and the delivery system design and controls. Included are guidelines for design, operation, maintenance, surveillance, and testing to provide for safe system and plant operation. Compliance with these guidelines will ensure that this system installation and operation will not produce a safety concern.

This 1987 Revision incorporates responses to NRC questions on a previous version of this document. The responses to the NRC questions have been incorporated throughout the text.

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Section 1

INTRODUCTION

1.1 SCOPE

This document sets forth design, construction, and operational guidelines for permanent hydrogen injection systems at boiling water reactors (BWRs) to facilitate the licensing process by reducing the case-by-case review and minimizing licensing efforts. As such, its purpose is to provide a reference document for utility use. NRC staff acceptance of these guidelines should minimize the amount of plant specific evaluations required.

The purpose of the hydrogen injection system is to inject hydrogen into the reactor coolant, presently via the feedwater system, to suppress the dissolved oxygen concentration. Suppressing the dissolved oxygen concentration and maintaining high purity in the reactor coolant will reduce the susceptibility of reactor piping and materials to intergranular stress corrosion cracking (IGSCC). This process is referred to as hydrogen water chemistry (HWC).

The scope of this document includes the currently available on-site hydrogen and oxygen gas supply options (i.e., compressed gas, cryogenic liquid, and electrolytic generation) and the gas delivery system design and controls. Included in this scope are the hydrogen injection system requirements for operation, maintenance, surveillance, safety precautions, and testing to provide for safe system and plant operation. Compliance with these requirements will ensure that the installation and operation of this system does not produce a safety hazard.

There are two primary regulatory concerns related to the permanent implementation of HWC: the potential impact of failures in the oxygen and hydrogen storage/handling systems on the plant safety systems and increased dose rates due to increased N-16 carry-over in the steam. For the oxygen and hydrogen storage/handling issue, this document addresses the possible failure modes of these systems. These failure modes include events external to these systems such as seismic, tornados, fire, vehicle hazards, etc. In addition, system internal events such as overpressurization and relief valve failures and the potential impact on plant structures and control room habitability are addressed. For these events, a mechanistic approach, as opposed to a probabilistic approach, is used as the basis for siting hydrogen and oxygen gas storage facilities. Using sufficiently conservative assumptions, the minimum distance between the hydrogen and oxygen supply facilities and safety-related structures is prescribed.

Injection of hydrogen into the feedwater system of a BWR can result in up to an approximate fivefold increase in the activity in the steam. Consequently, HWC will result in a minor increase in the site personnel exposure. However, over the life of the plant, HWC offers the potential for significantly reduced exposures because of the avoidance of recirculation pipe replacement and reduced pipe crack repair and inspection. This document provides recommendations to minimize the radiological impact of permanent HWC installations and to maintain exposures as-low-as-reasonably achievable (ALARA). In addition, the justification for increasing the main steam line radiation monitor set point to accommodate HWC is provided.

Some potential issues that will not impact continued safe plant operation but are associated with permanent HWC programs are:

1. Materials impact
2. Fuel impact
3. Reactor physics impact
4. Equipment qualification impact

Based on the conclusions of HWC laboratory testing and field testing at Dresden 2, there is no significant concern with hydrogen embrittlement. Based on the destructive examination of fuel exposed to HWC at Dresden 2, no significant impact on fuel performance is expected. Although the dissolved hydrogen concentration in the core inlet water increases slightly, the impact on core reactivity is insignificant, and reactor physics will not be affected. With regards to equipment qualification, dose rates inside the drywell close to the recirculation piping will decrease due to the increased carry-over of N-16 in the steam. Outside the drywell, the increase in the dose rates is relatively small relative to the integrated dose assumed for qualification tests.

1.2 BACKGROUND

The recirculating coolant in BWRs is high-purity (no additive) neutral pH water containing radiolytically produced dissolved oxygen (100-300 ppb). This level of dissolved oxygen is sufficient to provide the electrochemical driving force needed to promote IGSCC of sensitized austenitic stainless steel piping and similar structural components if the other two prerequisites for IGSCC (a sensitized microstructure (chromium depletion at the grain boundaries) and a tensile stress above the yield stress) are also present.

A variety of IGSCC remedies have been developed and qualified which address the sensitization and tensile stress aspects of stress corrosion cracking. Another approach for suppressing IGSCC involves modifying the BWR coolant environment to reduce the electrochemical driving force for IGSCC.

The HWC technique consists of reducing the coolant dissolved oxygen level from the present ~200 ppb to that level which, in combination with high water quality, has been shown to result in IGSCC immunity. The reduction in coolant oxygen is accomplished by the addition of hydrogen and the conductivity of the coolant is reduced (if needed) by improved water quality operational practices. The feasibility of suppressing oxygen by this approach has been demonstrated in short-term demonstrations in eight BWRs. A long-term verification test, which will extend over two or three 18-month fuel cycles, was initiated at Dresden-2 in April 1983.

An extensive laboratory investigation of the material performance consequences of combining oxygen suppression with conductivity control has demonstrated that substantial mitigation and possibly complete suppression of IGSCC can be achieved in ~280°C water with less than 20 ppb dissolved oxygen content if the conductivity was maintained below about 0.3 μs/cm. Results of slow strain rate tests at Dresden-2 have confirmed the anticipated improvement in the IGSCC resistance of sensitized austenitic stainless steel under HWC conditions and also supported other laboratory data indicating that HWC is a more innocuous service environment for most BWR plant structural materials than the non-HWC environment.

1.3 PREIMPLEMENTATION TESTING

Each utility should consider verifying the feasibility of implementing hydrogen water chemistry at their particular site.

In order to implement HWC, each utility should maintain water quality consistent with the "BWR Hydrogen Water Chemistry Guidelines" (1). This may result in an additional burden to the radioactive waste system. Each utility should evaluate the effect maintaining high water quality will have on plant systems.

Each utility should determine the hydrogen addition rate where IGSCC is mitigated. As the hydrogen addition rate is incrementally increased, the reactor dissolved oxygen concentration and the reactor recirculation piping electrochemical potential (ECP) decrease. Measuring the ECP of metal samples exposed to reactor recirculation water is one method for determining the point at which IGSCC immunity has been reached.

After determining the HWC operating parameters, verification methods such as Constant Extension Rate Testing (CERT) and Crack Arrest Verification (CAV) systems can be implemented.

Implementing HWC will increase the N-16 carry-over of the steam which causes increased on-site and off-site dose rates. These radiological impacts should be evaluated for acceptability. Section 6.1.3 of this report provides guidance for evaluating this impact and some techniques for mitigation.

As the hydrogen addition rate is increased, the feedwater dissolved oxygen concentration is reduced. During preimplementation testing, this parameter should be monitored. If the feedwater dissolved oxygen concentration is found to be unacceptably low, feedwater oxygen injection can be used to resolve this concern.

The guidelines for short-term HWC preimplementation testing to determine the hydrogen flow rate are not in the scope of this document. Also, system availability and other issues that are required to obtain licensing credit for HWC (e.g., reduced in-service inspection) are not addressed.

1.4 REFERENCE

1. "BWR Hydrogen Water Chemistry Guidelines: 1987 Revision." NP-4947-SR-LD. Palo Alto, Calif.: Electric Power Research Institute, to be published.

Section 2

GENERAL SYSTEM DESCRIPTION

Figure 2-1 shows the hydrogen addition system in simplified form. For this report, the system is divided into hydrogen supply, oxygen supply, hydrogen injection, and oxygen injection systems.

Options for hydrogen supply are discussed briefly below, and detailed descriptions of the main options are provided in section 3. Oxygen supply is also described in section 3. The gas injection systems are described in this chapter. Also described in this chapter are instruments and controls applicable to the entire system.

2.1 GENERAL DESIGN CRITERIA

The hydrogen water chemistry system is not safety-related. Equipment and components need not be redundant (except where required to meet good engineering practice), seismic category I, electrical class IE, or environmentally qualified. Nevertheless, proximity to safety-related equipment or other plant systems requires special consideration in the design, fabrication, installation, operation and maintenance of hydrogen addition system components. Section 9 of this document delineates the quality assurance and quality control requirements to assure a safe and reliable hydrogen addition system. In some cases these requirements are over and above those which are normally required for nonsafety-related installations.

The hydrogen addition system should suppress the dissolved oxygen concentration in the recirculation water to a point where IGSCC immunity is maintained at all reactor power levels at which the hydrogen addition system is operating.

2.2 HYDROGEN SUPPLY OPTIONS

Hydrogen can be supplied from three sources: (1) a commercial hydrogen supplier; (2) onsite production from raw materials; or (3) recovery and recycle of hydrogen from the off-gas system. Any combination of these three methods may, in principle, be appropriate at a given facility.

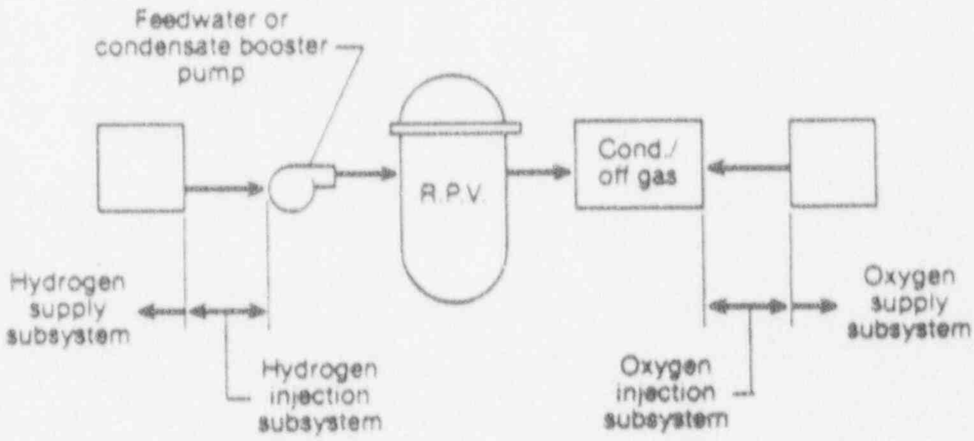

Figure 2-1. Hydrogen addition system.

2.2.1 Commercial Suppliers

Hydrogen can be obtained commercially from two types of sources: (1) merchant producers (i.e., companies that make hydrogen for the purpose of selling it to others) and (2) by-product producers (i.e., companies that produce hydrogen only as a by-product of their main business).

Hydrogen obtained in this manner is supplied as a high pressure gas or as a cryogenic liquid. The selection of gaseous or liquid supply options depends on system requirements such as flow rates and injection pressures and on-site considerations such as available separation distances and building strengths. In general, gaseous storage is preferred for low flow rates and small separation distances. Detailed considerations for gaseous and liquid hydrogen supply facilities are described in sections 3.1 and 3.2 of this report, respectively. Safety considerations are discussed in sections 4.1 and 4.2.

2.2.2 On-Site Production

Industrial processes for hydrogen production can be divided into two groups: electrolysis of water and thermochemical decomposition of a feedstock that contains hydrogen.

Detailed considerations for onsite production of hydrogen by electrolysis are described in section 3.3 of this report.

All other processes for producing high purity hydrogen involve thermochemical decomposition of hydrogen-containing feedstocks followed by a series of chemical and/or physical operations that concentrate and purify the hydrogen. While these processes are feasible, in principle, they are not currently envisioned for implementation. Therefore, these processes are not addressed in this report.

2.2.3 Recovery

Many processes are commercially available for separating, concentrating, and purifying hydrogen from refinery or by-product streams or for upgrading the purity of manufactured hydrogen. Processes are also being developed for the recovery and storage of hydrogen by the formation of rechargeable metal hydrides.

Although recovery of hydrogen is a viable option, near-term implementation of this option is not envisioned. Therefore, this option is not addressed in this report.

2.3 GAS INJECTION SYSTEMS

2.3.1 Hydrogen Injection System

The hydrogen injection system includes all flow control and flow measuring equipment and all necessary instrumentation and controls to ensure safe, reliable operation.

2.3.1.1 Injection Point Considerations. Hydrogen shall be injected at a location that provides adequate dissolving and mixing and avoids gas pockets at high points. Experience has shown that injection into the suction of feedwater or condensate booster pumps is feasible.

Injection into feedwater pumps will require hydrogen at high pressures (e.g., 150-600 psig). This may require either a compressed gas supply, compressors or a cryogenic hydrogen pump, depending on the supply option chosen. In the case of a liquid hydrogen storage system, this can also affect the sizing of the liquid hydrogen tank.

There may be pressure fluctuations in feedwater systems, depending on reactor power level and pump performance. The hydrogen addition system shall be designed to accommodate the full range of such fluctuations.

2.3.1.2 Codes and Standards. This system shall be designed and installed in accordance with OSHA standards in 29 CFR 1910.103.

Piping and related equipment shall be designed and fabricated to the appropriate edition of ANSI B31.1 or B31.3 for pressure-retaining components. Storage containers, if used, shall be designed, constructed, and tested in accordance with appropriate requirements of ASME B&PV Section VIII or API Standard 620. All components shall meet all the mandatory requirements and material specifications with regard to manufacture, examination, repair, testing, identification and certification.

All welding shall be performed using procedures meeting requirements in AWS D1.1, ANSI B31.1 or B31.3, or ASME B&PV, Section IX, as appropriate.

Inspection and testing shall be in accordance with requirements in ANSI B31.1, ANSI B31.3, or API 620, as appropriate.

System design shall also conform with pertinent portions of NUREG-0800, 10CFR50.48, Branch Technical position BTP CMEB 9.5-1, and appropriate standards and regulations referenced in this document. Appendix A provides a list of codes, standards, regulations, and published good engineering practices applicable to permanent hydrogen water chemistry installations. Each utility is responsible for identifying additional plant-specific codes and standards that may apply, such as State-imposed requirements, Uniform Building Code, ACI or AISC standards.

Piping and equipment shall be marked or identified in accordance with ANSI Z35.1.

2.3.1.3 <u>System Design Considerations</u>. Hydrogen piping from the supply system to the plant may be above or below ground. Piping below ground shall be designed for cathodic protection (or be coated and wrapped), the appropriate soil conditions such as frost depth or liquefaction, and expected vehicle loads. Guard piping around hydrogen lines is not required; however, consideration shall be given to its use for such purposes as protection from heavy traffic loads, leak detection and monitoring, or isolation of the potential hazard from nearby equipment, etc. All hydrogen piping should be grounded and have electrical continuity.

Excess flow valves should be installed in the hydrogen line at appropriate locations to restrict flow out of a broken line. Excess flow protection shall be designed to ensure that a line break will not result in an unacceptable hazard to personnel or equipment (BTP CMEB 9.5-1). The design features for mitigating the consequences of a leak or line break must perform their intended design function with or without normal ventilation.

Individual pump injection lines shall contain a check valve to prevent feedwater from entering the hydrogen line and to protect upstream hydrogen gas components. Automatic isolation valves should be provided in each injection line to prevent hydrogen injection into an inactive pump.

Purge connections shall be provided to allow the hydrogen piping to be completely purged of air before hydrogen is introduced into the line. Nitrogen or another inert gas shall be used as the purge gas. Gases shall be purged to safe locations, either directly or through intervening flow paths, such that personnel or explosive hazards are not encountered and undesirable quantities of gas are not injected into the reactor.

Area hydrogen concentration monitors are an acceptable way to ensure that hydrogen concentration is maintained below the flammable limit. If used, such monitors should be located at high points where hydrogen might collect and/or above use points that constitute potential leaks. Good engineering practice for locating hydrogen detector heads is to take into consideration the positive buoyancy of gaseous hydrogen. Detector heads shall be located so that the monitors shall be capable of detecting hydrogen leaks with or without normal ventilation. Each utility shall evaluate its particular system design and identify specific points where hydrogen concentration monitors should be installed. Examples of such points include flanged in-line devices (such as calibration spool pieces associated with mass flowmeters), outlets of purge/vent paths, or the items discussed in the following paragraph. Sleeves or guard pipes can be used as an alternative method to mitigate the consequences of a line break.

A hydrogen addition system will increase the hydrogen concentration in the feedwater, reactor, steam lines and main condenser. Each of these systems shall be reviewed for possible detrimental effects. A discussion of possible concerns is presented below.

1. Main Condenser

   The main condenser presently handles combustible gases. The hydrogen addition system does not significantly change the concentration or volume of noncondensables. Therefore, it is not anticipated that hydrogen addition will affect operation of the main condenser.

2. Off-Gas System

   Oxygen shall be added into the off-gas system to recombine with the hydrogen flow thus limiting the extent of the system handling hydrogen rich mixtures and reducing volumetric flow rates. The net effect will probably be a revised heat input into the recombined off-gas. The capability of the off-gas system to handle this revised heat load must be evaluated to ensure that temperature limits are not exceeded. Considerations in the design of the off-gas oxygen injection system should include loss of oxygen and runaway oxygen injection.

3. Steam Piping and Torus

   Hydrogen water chemistry may slightly increase the rate of hydrogen leakage into the torus via the safety relief valves. However, the rate of oxygen leakage will be decreased. Thus, the possibility of forming a combustible mixture is not significantly increased when compared to non-HWC operation.

4. Sumps

There are three water systems that may be affected by HWC: main condenser condensate, feedwater and reactor water. For sumps, which receive water from any of these three sources, the average hydrogen concentration in the water may increase slightly. The maximum expected concentration of hydrogen in the sump atmosphere should be determined to ensure that the hydrogen concentration remains below the lower combustible limit of hydrogen in air.

2.3.2 Oxygen Injection System

The oxygen injection system injects oxygen into the off-gas system to ensure that all excess hydrogen in the off-gas stream is recombined. It includes all necessary flow control and flow measurement equipment.

2.3.2.1 Injection Point Consideration. Oxygen should be injected into a portion of the off-gas system that is already diluted such that the addition of oxygen does not create a combustible mixture. If this is not possible, other system design considerations shall be provided in plant-specific cases to reduce the chances for off-gas fires.

2.3.2.2 Codes and Standards. The system shall be designed and installed in accordance with OSHA standards in 29 CFR 1910.104, and CGA G4.4, Industrial Practices for Gaseous Oxygen Transmission and Distribution Piping Systems.

Piping and related equipment shall be designed, fabricated, tested and installed in accordance with the appropriate edition of ANSI B31.1 or ANSI B31.3. Additional guidance on materials of construction for oxygen piping and valves is given in section 3.4 of this report, and in ANSI/ASTM G63, "Evaluating Nonmetallic Materials for Oxygen Service."

Welding shall be performed using procedures meeting requirements of AWS D1.1 or ASME B&PV, Section IX, as appropriate.

Piping shall be marked or identified in compliance with ANSI Z35.1.

System design shall also conform with appropriate NFPA, CGA, and other standards and regulations referenced elsewhere in this document. Each utility is responsible for identifying plant-specific codes and standards that may apply, such as State-imposed requirements, Uniform Building Code, ACI or AISC standards.

2.3.2.3 Cleaning. All portions of the system that may contact oxygen shall be cleaned as described in section 3.4 of this report, and in accordance with CGA G-4.1, Cleaning Equipment for Oxygen Service.

2.4 INSTRUMENTATION AND CONTROL

This subsection discusses the instrumentation, controls, and monitoring associated with the hydrogen addition system.

The instrumentation and controls include all sensing elements, equipment and valve operating hand switches, equipment and valve status lights, process information instruments, and all automatic control equipment necessary to ensure safe and reliable operation. Table 2-1 lists the recommended trips of the hydrogen addition system. The instrumentation shall provide indication and/or recording of parameters necessary to monitor and control the system and its equipment. The instrumentation shall also indicate and/or alarm abnormal or undesirable conditions. Table 2-2 lists the recommended instrumentation and functions. This table also includes instrumentation for hydrogen and oxygen supply options. Additional information on instrumentation and controls is provided in section 3.

System instrumentation and controls shall be centralized where feasible to facilitate ease of control and observation of the system. As a minimum, there shall be a system trouble alarm and/or annunciator provided in the main control room.

2.4.1 Hydrogen Injection Flow Control

Parallel flow control valves should be provided in the hydrogen injection line for system reliability and maintainability. If flow control is automatic, hydrogen flow rate should be controlled as a function of plant process parameters such as steam or feedwater flow.

The capability should be provided to adjust flow rate to each pump manually, if this is found to be necessary to achieve adequate hydrogen distribution.

Manual isolation valves shall be provided in each pump injection line to accommodate pump out-of-service conditions. Individual pump injection lines should contain automatic isolation valves interlocked to the corresponding pump, so that hydrogen is not injected into a pump that is not running.

Table 2-1

RECOMMENDED TRIPS OF THE HYDROGEN ADDITION SYSTEM

- Limiting low power level per plant safety analysis (Control Rod Drop Accident), if required by Tech Specs
- Reactor SCRAM
- Operator request (manual)
- Low residual oxygen in off-gas
- High residual oxygen in off-gas
- High area hydrogen concentration
- Low-oxygen injection system supply pressure or flow
- Off-gas train or recombiner train trip
- High hydrogen flow
- Differential hydrogen inlet and outlet of system*
- Oxygen concentration in hydrogen*
- Hydrogen concentration in oxygen*

*Electrolytic generation option.

Table 2-2

HYDROGEN ADDITION SYSTEM INSTRUMENTATION AND CONTROLS

Portion of Overall System	Parameter Measured or Function Performed	Record	Indicate	High Alarm	Low Alarm	Auto Control
Injection systems (H_2 and/or O_2)	Hydrogen flow (L, G, E)	(X)	X	(X)		Trip* on high flow
	Oxygen flow (L, G)	(X)	X			
	Off-gas residual oxygen (L, G, E)	(X)	X	X	X	Trip* on low oxygen Trip* on high oxygen
	Recirc water dissolved oxygen (L, G, E)	(X)	X	X		
	Area hydrogen concentration (L, G, E)		X	X		Trip*
	Hydrogen injection line pump interlock (L, G, E)					Isolate when pump is not in operation

*Trip of hydrogen/oxygen injection systems.

L = Liquid option.

G = Gaseous

E = Electrolytic option.

X = Required.

(X) = Recommended.

Table 2-2 (Continued)

HYDROGEN ADDITION SYSTEM INSTRUMENTATION AND CONTROLS

Portion of Overall System	Parameter Measured or Function Performed	Record	Indicate	High Alarm	Low Alarm	Auto Control
Hydrogen supply	Hydrogen storage tank level (L)		X		(X)	
	Hydrogen storage tank pressure gauge (L)		X			
	Hydrogen storage tank vacuum readout (L)		X			
	Hydrogen gas supply pressure (G)		X			
	Hydrogen gas storage temperature (G)		X			
	Differential flow rate (E)		X	X		Trip*
	Oxygen concentration in hydrogen (E)		X	X		Trip*
	Hydrogen concentration in oxygen (E)			X		Trip*
	Low temperature downstream of hydrogen vaporizer (L)		X		X	Trip H_2 Pump
	Hydrogen pump high discharge temperature (L)		X	X		Trip H_2 Pump
Oxygen supply	Oxygen tank level gauge (L)		X		(X)	
	Oxygen tank pressure gauge (L)		X			
	Oxygen tank vacuum readout connection (L)		X			
	Low temperature downstream of oxygen vaporizer (L)		X		X	

*Trip of hydrogen/oxygen injection systems

L = Liquid option.

G = Gaseous option.

E = Electrolytic option.

X = Required.

(X) = Recommended.

Provisions for shutoff of hydrogen injection shall be provided in the control room.

2.4.2 Oxygen Injection Flow Control

Parallel flow control valves should be provided in the oxygen injection line for system reliability and maintainability.

Oxygen flow rate shall be controlled to provide residual oxygen downstream of the recombiners. System controls shall be designed to ensure that oxygen injection continues after hydrogen flow stops, so that all free hydrogen is safely recombined.

2.4.3 Monitoring

Provision shall be made to monitor continuously the concentration of dissolved oxygen in the recirculation water. In obtaining samples of recirculation water for this purpose, appropriate containment isolation shall be provided in accordance with 10 CFR 50, Appendix A, General Design Criteria 3, 54, 55, 56, or 57.

Provision should be made to monitor continuously the concentration of oxygen and hydrogen in the off-gas flow downstream of the recombiners. Hydrogen and oxygen monitoring in the off-gas recombiner system should meet the acceptance criteria of Standard Review Plan 11.3 with the exception that automatic control functions are not required.

Section 3

SUPPLY FACILITIES

3.1 GASEOUS HYDROGEN

3.1.1 System Overview

Hydrogen gas can be supplied from either permanent high-pressure vessels or from transportable tube trailers. For the permanent storage system, gaseous hydrogen is stored in seamless ASME code vessels at pressures up to 2,400 psig and ambient temperatures. Transportable vessels are designed to DOT standards and store hydrogen at pressures up to 2650 psig at ambient temperatures. With either storage design, the gas is routed through a pressure control station which maintains a constant hydrogen supply pressure. In any event, the gaseous hydrogen system shall be provided by a supplier who has extensive experience in the design, operation and maintenance of associated storage and supply systems. Gaseous hydrogen shall be provided per CGA G-5 and G-5.3.

3.1.2 Specific Equipment Description

3.1.2.1 Hydrogen Storage Vessels. The hydrogen storage bank shall be composed of ASME Code gas storage vessels. Each tube shall be constructed as a seamless vessel with swagged ends. Specific tube design shall be based on ASME Unfired Pressure Vessel Code, Section VIII, Division 1, including Appendix XIV-70.

The tube bank shall be supported to prevent movement in the event of line failure and each tube shall be equipped with a close-coupled shutoff valve. As an alternative, one safety valve per bank of tubes can be used, provided the safety valve is sized to handle the maximum relief from all tubes tied into the valve. Each bank shall be equipped with a thermometer and a pressure gauge, as is necessary for proper filling.

3.1.2.2 Transportable Hydrogen Storage Vessel. Transportable hydrogen vessels shall be constructed, tested, and retested (every 5 years), in accordance with DOT specifications 3A, 3AA, 3AX, or 3AAX. All valving and instrumentation shall be identical to section 3.1.2.1.

3.1.2.3 Pressure Reducing Station. The pressure control station shall be of a manifold design. The manifold shall have two (2) full-flow parallel pressure reducing regulators. The discharge pressure range of these regulators shall be adjustable to satisfy plant hydrogen injection requirements. Pressure gauges shall be provided upstream and downstream of the regulators. Sufficient hand valves shall be provided to ensure complete operational flexibility.

An excess flow check valve shall be installed in the manifold immediately downstream of the regulators to limit the flow rate in the event of a line break. The stop-flow set point shall be determined by each plant and should be set between the maximum plant flow requirements and the full C_v of the flow control valves. Additional guidance on excess flow protection is provided in section 2.3.1.3.

3.1.2.4 Tube Trailer Discharge Stanchion. A tube trailer discharge stanchion shall be provided for gaseous product unloading. The stanchion shall consist of a flexible pigtail, shutoff valve, check valve, bleed valve, and necessary piping. Filling apparatus shall be separated from other equipment for safety and convenience, and protected with walls or barriers to prevent vehicular collision.

A tube trailer grounding assembly shall be provided for each discharge stanchion to ground the tube trailer before the discharge of hydrogen begins.

3.1.2.5 Interconnecting Pipeline. All equipment and interconnecting piping supplied with this system shall be installed in compliance with the following standards:

- American National Standards Institute (ANSI) B31.1, Power Piping, or B31.3, Chemical Plant and Petroleum Refinery Piping.
- National Fire Protection Association (NFPA) 70, National Electrical Code.
- NFPA-50A, Bulk Hydrogen Systems.
- All applicable local and national codes.

There are several suitable field installation techniques which are based on industrial experience. The following are guidelines which may be used for field connections:

- Copper-to-Copper, Brass-to-Brass, and Copper-to-Brass Socket Braze Joints.

--Silver Alloy	45% Ag, 15% Cu, 16% Zn, 24% Cd., ASTM B260-69T and AWS A5.8-69T, BAg-1 Melting Range-Solidus-607.2°C Liquidus-618.3°C
--Flux	Working Range 593.3°C to 871.1°C

- Copper, Brass, Carbon Steel, and Stainless Steel N.P.T. Threaded Joints.

--TEFLON* Tape**	SCOTCH*** Number 48 Tape** or equal . -195.5°C to +204.4°C, 0 to 3,000 psig. Wrapped in direction of threads.

- Flange Joints (On all Materials).

--Ring Gasket Material, Low Pressure (720 psig maximum)	Precut T.F.E. impregnated asbestos, 1/16 inch thickness. Garlock 900 or equal. -195.5°C to +168.3°C, 0 to 900 psig.
--Ring Gasket Material, High Pressure	FLEXITALLIC**** Type. Material to be 0.175 inch thick 304 stainless steel with TEFLON filler and 0.125 inch carbon steel guide ring.

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*TEFLON is a trademark of E. I. duPont de Nemours & Co., Wilmington, DE 19898.

**If tape is used, electrical continuity/grounding of each piping section should be confirmed.

***SCOTCH is a trademark of 3M Company, St. Paul, MN 55101.

****FLEXITALLIC is a trademark of Flexitallic Gasket Co., Bellmawr, NJ 08031.

--Antiseize Compound	For flange face, nut, and bolt lubrication. Halocarbon 25-5S grease or equal. -195.5°C to +176.6°C, 0 to 3,000 psig. DO NOT USE ON ALUMINUM, MAGNESIUM, OR THEIR ALLOYS UNDER CONDITIONS OF HIGH TORQUE OR SHEAR.

- Carbon Steel, Stainless Steel, and Aluminum Alloys Socket and Butt welds.

--Welding Procedure	Gas Metal Arc Welding (GMAW), Gas Tungsten Arc Welding (GTAW), Shielded Metal Arc Welding (SMAW), or Plasma Arc Welding (PAW); with appropriate filler material and shielding gas. Proper surface and joint preparation (in regard to cleaning and clearances) should be exercised.

3.1.2.6 Component Cleaning. All components that contact hydrogen must be free of moisture, loose rust, scale, slag, and weld spatter; they must be essentially free of organic matter, such as oil, grease, crayon, paint, etc. To meet these objectives, system components shall be cleaned in accordance with standard industrial practices, as recommended by the gas supplier, prior to and following system fabrication.

3.2 LIQUID HYDROGEN

3.2.1 System Overview

Liquid hydrogen is stored in a vacuum-jacketed vessel at pressures up to 150 psig and temperatures up to -403°F (saturated). Based on data relating hydrogen injection pressures to BWR plant power levels, hydrogen supply from a liquid source can be provided directly from a tank or pumped into supplemental gaseous storage. Gaseous storage requirements are identified in section 3.1. The required supply pressure shall be based on pressure requirements at the point of hydrogen injection and line losses from the hydrogen supply system to the injection point.

Feedwater pressure requirements and line losses must not exceed 120 psig if hydrogen is to be supplied directly from a liquid tank.

In any event, the liquid hydrogen system shall be provided by a supplier who has extensive experience in the the design, operation and maintenance of associated storage and supply systems, such as cryogenic pumping. Liquid hydrogen shall be provided in accordance with CGA G-5 and G-5.3.

3.2.2 Specific Equipment Description

3.2.2.1 Cryogenic Tank. Tanks for liquid hydrogen service are available with capacities between 1,500 gallons and 20,000 gallons. An "inner vessel" or "liquid container" is supported within an "outer vessel" or "vacuum jacket," with the space between filled with insulation and evacuated. Necessary piping connects from inside of the inner vessel to outside of the vacuum jacket. Gages and valves to indicate the control of hydrogen in the vessel are mounted outside of the vacuum jacket. Legs or saddles to support the whole assembly are welded to the outside of the vacuum jacket.

Inner vessels are designed, fabricated, tested, and stamped in accordance with Section VIII, Division 1 of the ASME Code for Unfired Pressure Vessels. Materials suitable for liquid hydrogen service must have good ductility properties at temperatures of -422°F per CGA G-5. The cryogenic operating temperatures of these vessels preclude material degrading mechanisms such as corrosion or hydrogen embrittlement. The constant operating vessel pressures assure that flaw growth due to cyclic stress loading will not occur. The inner vessel is subject to a required pressure test which insures that no flaws exist that could cause a failure at or below the set pressure of the vessel's redundant relief devices. In addition to ASME Code inspection requirements, 100% radiography of the inner vessel longitudinal welds shall be completed. The tank outer vessel shall be constructed of carbon steel and shall not require ASME certification.

Insulation between inner and outer vessels shall be either perlite, aluminized mylar, or suitable equal. Fibrous or blanket insulation, such as bonded glass fibers or rock wool, shall not be used because of the potential for liquid-saturated missiles which would occur only as a result of vessel failure. The annular space should be evacuated to a high vacuum of 50 microns or less.

Tank control piping and valving should be installed in accordance with ANSI B31.1 or B31.3. All piping shall be either wrought copper or stainless steel. The following tank piping subsystems shall be provided:

- Fill circuit, constructed with top and bottom lines so that the vessel can be filled without affecting continuous hydrogen supply.
- Pressure-build circuit, to keep tank pressures at operational levels.
- Vacuum-jacketed liquid fill and pump circuits, where applicable.

3.2.2.2 <u>Overpressure Protection System</u>. Safety considerations for the tank shall be satisfied by dual full-flow safety valves and emergency backup rupture discs. The primary relief system shall consist of two sets of a minimum of one (1) rupture disk and safety valve piped into separate "legs." Relief devices shall be connected in parallel with other relief devices. The system shall be coupled by a 3-way diverter valve or tie bar interlock so that one leg is opened when the other is closed. With this arrangement, a minimum of one safety valve and one rupture disk will be available at all times. The dual primary relief systems with 100% standby redundancy allows maintenance and testing to be performed without sacrificing the level of protection from overpressure.

The primary relief system shall comply with the provisions of the American Society of Mechanical Engineers (ASME) Pressure Vessel Codes and the Compressed Gas Association (CGA) Standards.

The tank shall also be supplied with a secondary relief system not required by the ASME Codes. This system shall be totally separate from the primary relief system. It shall consist of a locked open valve, a rupture disk, and a secondary vent stack. This rupture disk shall be designed to burst at 1.33 times maximum allowable working pressure (MAWP).

Supply system piping that may contain liquid and can be isolatable from the tank relief valves shall be protected with thermal relief valves. All outlet connections from the safety relief valves, rupture devices, bleed valves, and the fill line purge connections shall be piped to an overhead vent stack, per CGA G-5, Section 7.3.7.

Two relief devices shall be installed in the tank's outer vessel to relieve any excessive pressure buildup in the annular space.

Hydrogen tanks and delivery vehicles shall be grounded per CGA P-12, Sections 5.4.5 and 5.7.1.2. The storage system shall be protected from the effects of lightning per NFPA 78, Chapter 6.

Excess flow protection shall be added to the tank's liquid piping wherever a line break would release a sufficient amount of hydrogen to threaten safety-related structures. An acceptable methodology is identified in section 4.2.2, "Pipe Breaks."

3.2.2.3 Instrumentation. The tank shall be supplied with a pressure gauge, a liquid level gauge, and a vacuum readout connection. These gauges are sufficient for normal monitoring of the tank condition. Instrumentation for remote monitoring, such as high/low-pressure switches, pressure and level transmitters may be added. A listing of supply system instrumentation and control is identified in section 2.4.

3.2.2.4 Liquid Hydrogen Pump and Controls. The liquid hydrogen pump shall be of proven design to provide continuous hydrogen supply in unattended, automatic operation. The following items comprise the more important system controls.

3.2.2.4.1 Positive isolation valve. A positive isolation valve shall be used to control the liquid feed into the pumping system per NFPA 50B. The valve shall be a failed-closed, pneumatically operated valve. The valve shall only be open during pump operation, shall close in any fault mode, and shall be able to be remotely overridden in case of emergency.

3.2.2.4.2 System overpressure shutdown. Although the system is protected by safety relief valves and rupture discs, system overpressure shall be avoided by shutting down the pumps at high pressure.

3.2.2.4.3 Temperature indicating switch. A temperature switch shall continuously monitor the downstream gas line for low temperature and shall trip the liquid pump to protect downstream equipment from low temperatures.

3.2.2.4.4 Pump operation. Pump operation shall be continuously and automatically monitored. Operation which results in pump cavitation, high temperature at the pump discharge, or low temperature downstream of the vaporizer shall cause the pump to be shut down by the remote control panel. The fault shall be indicated on the remote control panel by an audible alarm and light indication.

3.2.2.4.6 Purging of controls. All electrical components in hydrogen service should be designed in accordance with NFPA 70. Only nitrogen or another inert gas shall be used for purging pump motors, control panels and valves.

3.2.2.5 Interface with Gaseous System. Liquid hydrogen pump systems typically require a gaseous storage system as a surge or back-up to plant hydrogen supply. These storage systems shall be designed in accordance with section 3.1, Gaseous Hydrogen. Whenever a gaseous backup is used in conjunction with a liquid hydrogen system, switchover controls shall be provided.

3.2.2.6 Vaporization. Vaporization of the liquid hydrogen shall be achieved by the use of ambient air vaporizers. Vaporizer design, installation and operation shall take guidance from NFPA 50A and 50B.

The vaporizer should feature a star fin design and aluminum alloy construction. For a combined liquid and gaseous storage system, the vaporizers used should have a design pressure consistent with plant injection pressure requirements. The units may be piped in parallel such that each unit can operate independently. Parallel vaporizer assemblies shall be sized for the peak hydrogen flow required for each plant and shall provide for periodic intervals for defrosting, as appropriate. Other atmospheric vaporization systems may be utilized if their capacity is demonstrated to be adequate for the plant flow and ambient conditions.

For a pumped liquid only storage system, the vaporizer must withstand maximum pressures generated from the cryogenic pump. These vaporizers shall be equipped with stainless steel lining designed to 3500 psig.

3.3 ELECTROLYTIC

3.3.1 System Overview

The disassociation of water by electrolysis is an acceptable method of obtaining the gases needed for hydrogen water chemistry. This can be done on site and the gases can conveniently be generated at the rate used. The electrolytic gas generator should be proven equipment, the same as used in other industrial applications. Depending on the generator operating pressure, either hydrogen compressors or pressure breakdown (control) is utilized to match plant hydrogen injection pressure requirements. The electrolytic system shall be provided by a supplier who has extensive experience in the design, operation and maintenance of these systems.

3.3.2 Specific Equipment Description

Equipment and processes associated with the electrolytic method of providing the HWC gases include rectifiers, the electrolytic cells, scrubbers, compressors, piping, valves and associated controls.

3.3.2.1 Gas Generator. Water is dissociated into hydrogen and oxyen in the electrolytic cells by the direct current electricity provided through the rectifiers. The water flows into the cells, at the rate dissociated, where it forms a solution with the electrolyte used to carry the electrical current from one electrode to the other. Hydrogen is formed at one electrode and oxygen at the other, which is dependent on current direction. The electrodes are separated by a membrane which is permeable for the electrolyte but which keeps the gas bubbles separate as they rise to the collection outlets of the cells.

3.3.2.2 Vessels. Unless exempted because of size (smaller than 120 gallons of water) or pressure (less than 15 psig), for industrial safety reasons, the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section VIII, Division 1, shall apply to the design and construction of vessels. The code design pressure and temperature shall be selected to be above the highest pressure and temperature that can be reached during operation.

3.3.2.3 Piping. Piping and related equipment shall conform to the American National Standard Code (ANSI) for Piping B31.1 or B31.3 except that nonmetallic materials may be used in low pressure applications if supported by experience and/or tests which have demonstrated their suitability for the service conditions, and operating pressure and temperature conditions are within the material manufacturer's specifications. Typical nonmetallic materials are limited to pressures below 150 psig and temperatures below 140°F.

3.3.2.4 Valves. Valves should be designed such that the prevention of hydrogen leakage into an enclosed area does not rely on a single packing. Valves in any portion of the hydrogen flow path that is at subatmospheric pressure should be designed for zero inleakage of air.

Welded connections shall be used on all hydrogen piping that may operate below atmospheric pressure.

The oxygen flow path, including valves, shall not contain greases, oils, or other combustible materials which can ignite at the proper conditions of temperature and velocity.

Valves in the hydrogen flow path in or downstream of any point where the pressure can be below atmospheric should have spark-resistant rubbing and impacting surfaces if the rubbing or impacting velocities can exceed the spark threshold.

A helium or soap bubble test as appropriate shall be performed to assure a leak tight system after installation.

3.3.2.5 Compressors. If a mechanical method of gas compression is employed, it should be located at the gas generation facilities. A gas pressurization method may be employed which does not require mechanical compressors and which permits gas generation at a rate equal to gas usage. However, if a mechanical compressor is used, it shall meet the following requirements:

(a) The pressure gradient at any seal should be outward whenever the compressor contains hydrogen.

(b) The shaft seal leakage shall not discharge into any enclosed space that is not either continuously purged with an inert gas or ventilated to avoid an explosive mixture of air and hydrogen assuming the greatest potential rate of shaft seal leakage.

(c) The compressor shall not introduce unacceptable levels of organics and/or fluorides/chlorides into the hydrogen.

(d) The compressor shall be designed to permit purging of all compartments before and after maintenance.

(e) The compressor should be dry lubricated and should be of the diaphragm type.

Where gas storage volumes are used, their size should be minimized. Where practical for the application, a type of compressor should be used that does not require surge tanks.

3.3.2.6 Gas Generator Shelter. Passive ventilation shall be provided for the gas generating room of the equipment shelter. Inlet openings shall be provided at floor level in exterior walls and outlet openings shall be located at the high point of the room. Inlet and outlet openings shall have an arrangement and sufficient area to assure fault-free passive ventilation. The discharge from outlet openings shall be directed to a location that has no ignition sources.

The gas generating room of the shelter shall be partitioned away from all other rooms that could contain ignition sources. The rectification equipment shall be partitioned away from the gas generation equipment.

Equipment for space heating of the gas generating room shall not contain any ignition sources and shall not allow gases, including air, to pass out of the room to an ignition source in a heating system.

Windows and doors shall be in exterior walls only. Windows shall be made of shatterproof glass or plastic in metal frames.

The shelter shall be of noncombustible materials (except for the transparent materials used in windows).

3.4 LIQUID OXYGEN

3.4.1 System Overview

Liquid oxygen is stored in a vacuum-jacketed vessel at pressures up to 250 psig and temperatures up to -251°F (saturated). Oxygen taken from the vessel shall be vaporized through ambient air vaporizers and routed through a pressure control station which maintains gas pressures within the desired range. The liquid oxygen system shall be provided by a supplier who has extensive experience in the design, operation and maintenance of associated storage and supply systems. Liquid oxygen shall be provided per CGA G-4 and G-4.3.

3.4.2 Specific Equipment Description

3.4.2.1 Cryogenic tank. Tanks for liquid oxygen service, with capacities between 3,000 gallons and 11,000 gallons are similar in principle. An "inner vessel" or "liquid container" is supported within an "outer vessel" or "vacuum jacket," with insulation provided in the space between the tanks. Necessary piping connects from inside of the inner vessel to outside of the vacuum jacket. Gages and valves to indicate the control of product in the vessel are mounted outside of the vacuum jacket. Legs or saddles to support the whole assembly are welded to the outside of the vacuum jacket.

Inner vessels shall be designed, fabricated, tested and stamped in accordance with Section VIII, Division 1, of the ASME Code for Unfired Pressure Vessels. Materials suitable for liquid oxygen service must have good ductility properties at cryogenic temperatures of -320 F per CGA G-4. The outer vessel should be constructed of carbon steel and does not require ASME certification.

Insulation between inner and outer vessels shall be either perlite, aluminized mylar or suitable equal. The annular space should be evacuated to a high vacuum of 50 microns or less.

Tank control piping and valving should be installed in accordance with ANSI B31.1 or B31.3. All piping shall be either wrought copper or stainless steel. The following tank piping subsystems shall be provided:

- Fill circuit constructed with top and bottom lines so that the vessel can be filled without affecting system operation.
- Pressure-build circuit, to keep tank pressures at operational levels.
- Economizer circuit, to preferentially feed oxygen gas from vessel vapor space to process.

3.4.2.2 Overpressure Protection System: Safety considerations for the tank shall be satisfied by dual full-flow safety valves and emergency backup rupture discs. The primary relief system shall consist of two sets of one (1) safety valve and one (1) rupture disc piped into separate legs, coupled by a three-way valve. This dual primary relief system with 100% standby redundancy allows maintenance and testing to be performed without sacrificing the level of protection from overpressure.

The primary relief system shall comply with the provisions of the ASME Pressure Vessel Codes and the Compressed Gas Association (CGA) Standards.

Annular space safety heads shall be provided to relieve any excess positive pressure buildup which might result from a leak in an inner vessel. Supply system piping that may contain liquid and can be isolatable from the tank relief valves shall be protected with thermal relief valves.

The tank shall be supplied with a pressure gauge, a liquid level gauge, and a vacuum readout connection. These gauges are sufficient for normal monitoring of the tank condition. Instrumentation for remote monitoring, such as high/low-pressure switches, pressure and level transmitters may be added. A listing of supply system instrumentation and control is identified in section 2.4.

3.4.2.3 Vaporization. The vaporization of the liquid oxygen shall be achieved by the use of ambient air vaporizers.

The vaporizer should feature a star fin design and extruded aluminum alloy construction. The vaporizers shall have a minimum design pressure of at least 300 psig. The units shall be piped in parallel such that each unit can operate independently. Parallel vaporizer assemblies shall be sized to handle peak plant flow requirements and shall provide for periodic intervals for defrosting, as appropriate. Other atmospheric vaporization systems may be utilized if their capacity is demonstrated to be adequate for the plant flow and ambient conditions.

3.4.2.4 Pressure Control Station. The pressure control station shall be of a manifold design. The manifold shall have two (2) full-flow parallel pressure reducing regulators. The discharge pressure range of these regulators shall be adjustable to satisfy plant oxygen injection requirements. Pressure gauges shall be provided upstream and downstream of the regulators and sufficient hand valves shall be provided to ensure complete operational flexibility.

Protection of downstream equipment from low-oxygen temperatures shall be included in the system design.

3.4.3 Materials of Construction for Oxygen Piping and Valves

The design and installation of oxygen piping and related equipment shall be in accordance with ANSI B31.1 or B31.3 and the following guidelines for material selection for oxygen systems.

Observations of past oxygen fires indicate that ignition can occur in carbon steel and stainless steel piping systems operating at, or near, sonic velocity. Friction from high velocity particles is considered to be the source of ignition. Copper, brass, and nickel alloys have the characteristic of melting at temperatures below their respective ignition temperatures. This makes these materials extremely resistant to ignition sources, and once ignited, they exhibit a much slower rate of burning than carbon or stainless steels.

As a result of these observations, the following materials, in order of preference, are acceptable for oxygen service. In the case of carbon steel or stainless steel, the maximum velocity of gaseous oxygen shall be within guidelines established by the Compressed Gas Association CGA Pamphlet CGA-4.4, "Industrial Practices for Gaseous Oxygen, Transmission and Distribution Piping Systems."

- Copper
- Brass
- Monel
- Stainless Steel
- Carbon Steel

If steel pipe is to be used for the system and some local flow conditions could cause the velocity to exceed that established in CGA G-4.4, then that portion of the system must be converted to a copper-based alloy and extend a minimum of 10 diameters downstream of the point of return to the allowable velocity. These local flow conditions may occur at control valves, orifices, branch line take-off points, and in the discharge piping of safety relief devices.

Valves that open rapidly are not suitable for oxygen service, since rapid filling of an oxygen line will result in a temperature increase due to adiabatic compression. As a result of this phenomenon, ball valves and automatic valves may only be used with the following restrictions:

- Valve bodies shall be made of a copper alloy. Balls shall be monel or brass. Valve seats and seals should be teflon, non-plasticized Kel-F, Kalrez, or Viton.
- Ball valves may not be used as process control valves in throttling or regulating service. Ball valves may be used as isolation valves, emergency shutoff valves, or vent or bleed valves where they are either fully open or fully closed.
- Pneumatic or electric ball valves used for on-off services shall have an actuation time from fully closed to fully open of 4 seconds or greater for pressures up to 250 psig. No restriction is placed on actuation time from fully open to fully closed. Piping immediately downstream must be a straight run of copper-bearing material for a minimum of 10 diameters.
- Pneumatic or electric ball valves used for emergency service may be fully open or fully closed to the emergency position, with no restrictions on actuation time.

Suitable valve packing, seats, and gasket materials are listed below in order of preference from the oxygen compatibility basis only.

- Teflon
- Glass-filled Teflon
- Nonplasticized Kel-F
- Garlock 900
- Viton or Viton A

3.4.4 Oxygen Cleaning

All piping, fittings, valves, and other material which may contact oxygen shall be cleaned to remove internal organic, inorganic, and particulate matter in accordnace with CGA 4.1. Observation has shown that ignition can occur in properly designed piping systems when foreign matter is introduced. Therefore, removal of contaminants such as grease, oils, thread lubricants, dirt, water, filings, scale, weld spatter, paints, or other foreign material is essential. Cleaning should be accomplished by precleaning all parts of the system, maintaining cleanliness during construction, and by completely cleaning the system after construction.

Section 4

SAFETY CONSIDERATIONS

4.1 GASEOUS HYDROGEN

4.1.1 Site Characteristics of Gaseous and Liquid Hydrogen

4.1.1.1 Overview. Review of the following site characteristics shall be conducted by each BWR facility in locating the gaseous and/or liquid hydrogen supply systems:

1. Location of supply system in proximity to exposures as addressed in NFPA 50A and 50B.
2. Route of hydrogen delivery on site.
3. Location of supply system in proximity to safety-related equipment.

4.1.1.2 Specific Considerations.

4.1.1.2.1 Fire protection. The area selected for hydrogen system siting shall meet or exceed all requirements for protection of personnel and equipment as addressed in NFPA 50A and 50B, gaseous and liquified hydrogen systems, respectively. Each standard identifies the maximum quantity of hydrogen storage permitted and the minimum distance from hydrogen systems to a number of exposures.

The need for additional fire protection for other than the hydrogen facility shall be determined by an analysis of local conditions of hazards on-site, exposure to other properties, water supplies, and the probable effectiveness of plant fire brigades in accordance with NFPA 50A and 50B.

4.1.1.2.2 Security. All hydrogen storage system installations shall be completely fenced, even when located within the owner-controlled area. Lighting shall be installed to facilitate night surveillance.

4.1.1.2.3 Route of hydrogen delivery on site. Each plant should determine the route to be taken by hydrogen delivery trucks through on-site and off-site areas. In order to protect the hydrogen storage area from any vehicular accidents, truck barriers shall be installed around the perimeter of the system installation.

Within the plant security area, all deliveries shall be controlled per the requirements of 10 CFR 73.55.

4.1.1.2.4 Location of storage system to safety-related structures. Each plant shall determine that the location of the hydrogen storage system is acceptable relative to safety-related structures and equipment considering the hazards described in sections 4.1.2, 4.1.3, 4.2.1 and 4.2.2.

4.1.2 Gaseous Storage Vessel Failure

Gaseous storage vessels in the scope of this report are the commercially available, seamless, swagged-ended vessels that are commonly referred to as "hydril tubes." This section addresses the non-mechanistic rupture failure of single vessels and the separation distances required to avoid damage to safety-related equipment. Simultaneous failure of multiple vessels is not addressed because the inherent strength of the vessel makes them unsusceptible to failure from outside forces. These vessels shall be capable of withstanding tornado missiles (NUREG-0800) and site specific seismic loading due to horizontal and vertical accelerations acting simultaneously.

These features eliminate common cause vessel failures so that the maximum postulated instantaneous release is the fully pressurized contents of the largest single vessel. The potential consequences of such a release, a fireball or an explosion, are addressed in order.

4.1.2.1 Fireball. The thermal flux versus distance from the fireball center are shown on figure 4-1 for the two most common vessel sizes. These fluxes and durations will not adversely affect safety-related structures. However, each utility shall review any unique site characteristics to assure all safety-related equipment will function in the event of a fireball.

4.1.2.2 Explosion. When a gaseous storage vessels ruptures, the expansion of the high-pressure gas results in rapid turbulent mixing with the surrounding air. In

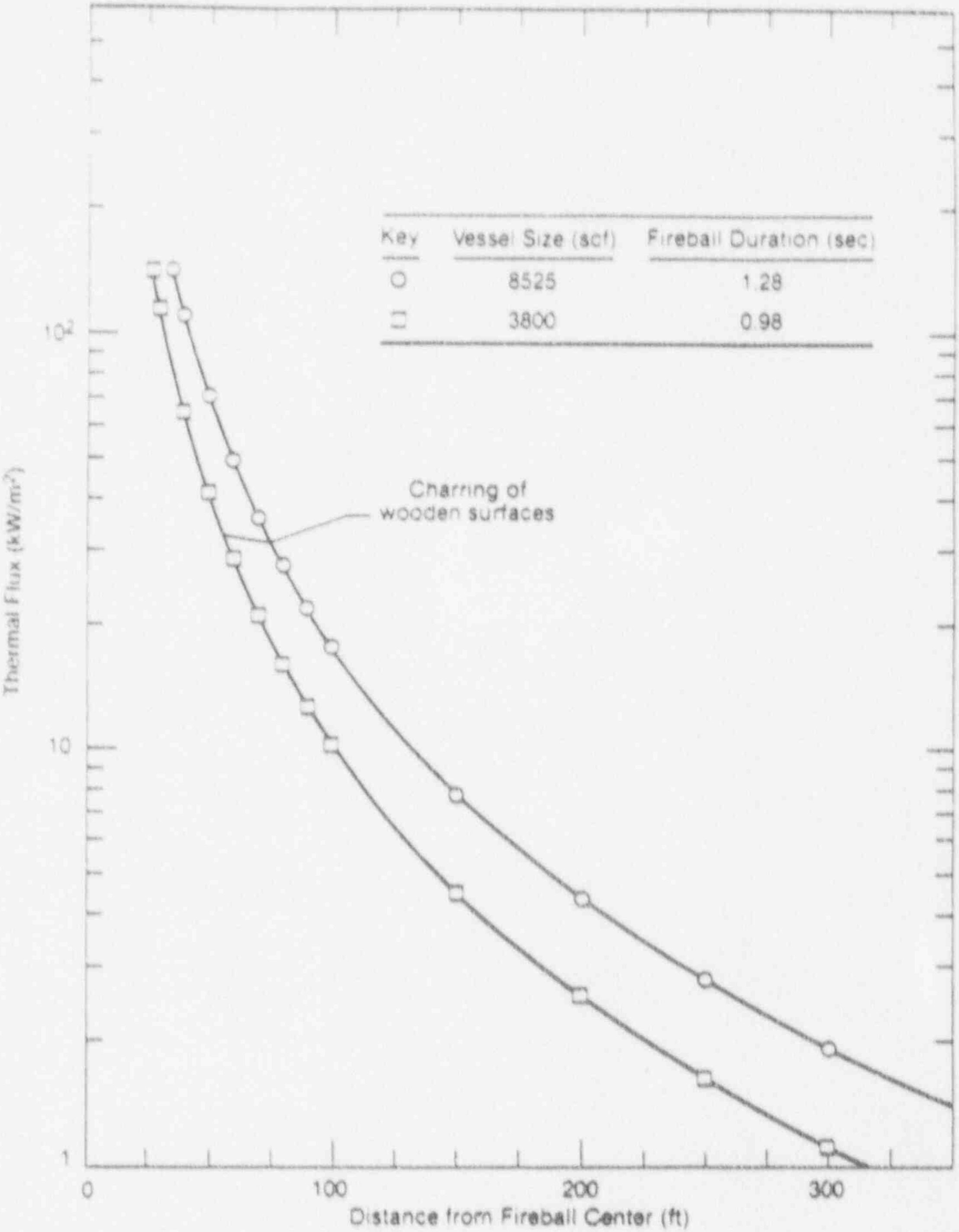

Figure 4-1. Thermal flux vs. distance from fireball center for gaseous hydrogen storage system.

the case of gaseous hydrogen, the release will go through the detonation limits of 18.3 - 59% before the wind can translate the mixture. Consequently, any explosion blastwaves will originate at the vessel rupture site. For this report, it is conservatively assumed that 100% of the vessel contents will contribute to the blastwave and that the TNT-hydrogen equivalence is 20% on an energy basis (520% on a mass basis). This translates to 27.1 lbs. of TNT per 1000 standard cubic feet (SCF) of gaseous hydrogen. Using this conversion factor and U.S. Army Technical Manual TM5-1300, blast overpressures and impulses can be calculated as functions of distance from the vessel location. These blast parameters could then be compared to the dynamic strength of safety-related structures.

An evaluation entitled "Separation Distances Recommended for Hydrogen Storage to Prevent Damage to Nuclear Power Plant Structures From Hydrogen Explosion" was performed for EPRI by R. P. Kennedy. This evaluation, which is included as appendix B of these guidelines, recommends separation distances based on quantities of stored hydrogen and building design factors. The recommendations are provided in the form of step-by-step procedures, with subsequent steps requiring additional work but resulting in reduced distances from the previous step. The procedure to determine acceptable separation distances is outlined below.

1. For any reinforced concrete or masonry walls at least 8-inches thick, the upper curve on figure 4-2 provides conservative separation distances as a function of vessel size. If this is acceptable, then no further work is needed. Otherwise, proceed to step 2.

2. For reinforced concrete walls at least 18-inches thick, with known static strength and percent tensile rebar, eq. 7 in appendix B can be used to determine required separation distances. The two lower curves on figure 4-2 are representative examples of design parameters for walls of nuclear power plants. Walls with different parameters should be analyzed using the methods in appendix B, pages 10 through 13. If this is acceptable, then no further work is needed. Otherwise, proceed to step 3.

3. For separation distances closer than allowed by the above 1 and 2, perform a dynamic blast capacity analysis in accordance with NUREG/CR-2462 (1).

For all storage locations, the vessel(s) and the foundation(s) shall be designed to remain in place for both design-basis tornado characteristics and site-specific flood conditions.

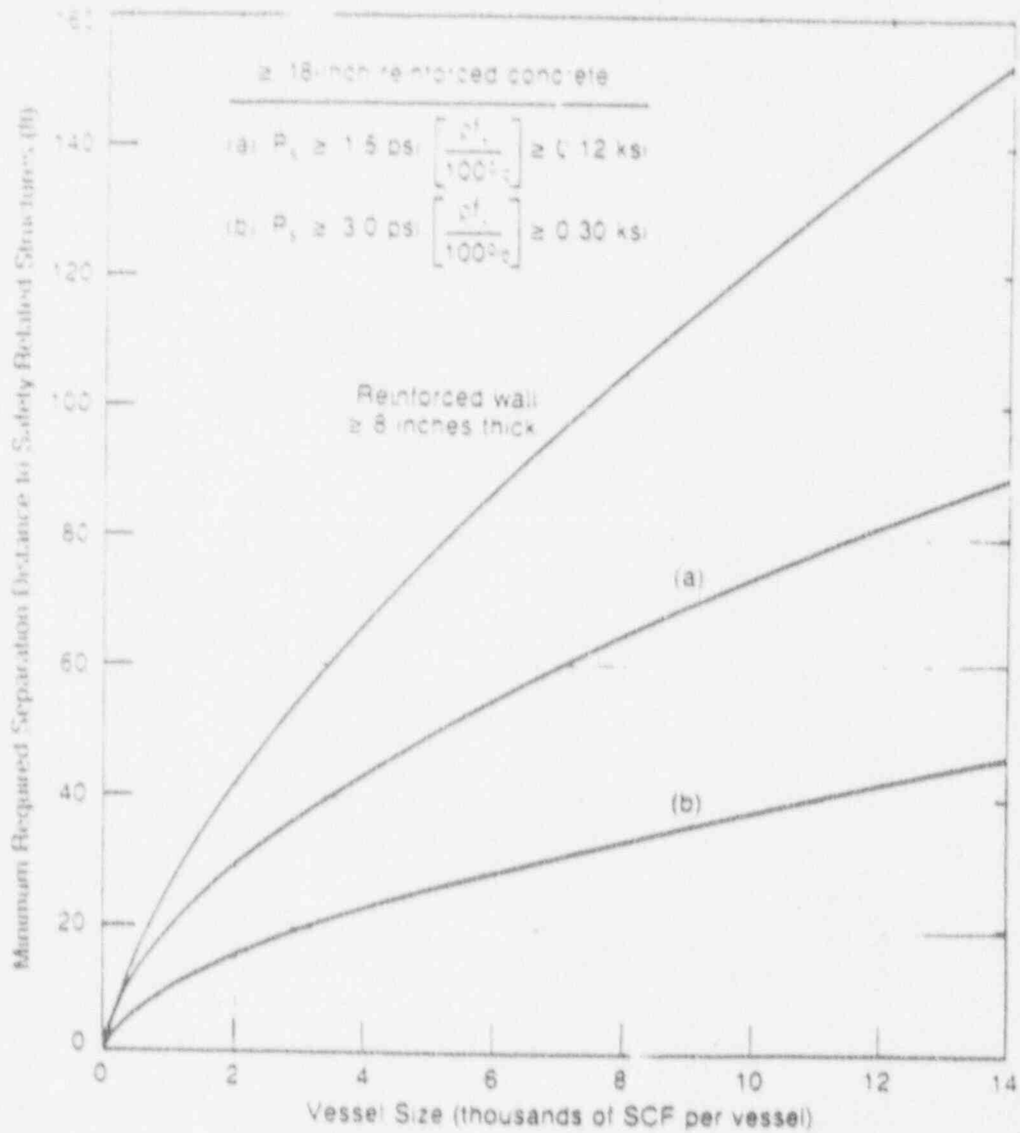

Figure 4-2 Minimum required separation distances to safety-related structures versus vessel size for gaseous hydrogen storage system.

4.1.3 Gaseous Pipe Breaks

This section addresses the requirements for hydrogen piping systems attached to gaseous storage vessels up to the point where excess flow protection is provided. The criteria for acceptable siting for the event of a pipe break are:

1. Dilution of resultant release below the lower flammability limit of 4% before reaching air pathways into safety-related structures.

2. Minimum separation distances for the blast damage criteria outlined in section 4.1.2.

It is conservatively assumed that all releases occur while the storage vessel is at 2,450 psig. This is the maximum allowable working pressure of the majority of commercially available vessels.

Gaseous releases at elevated pressures result in supersonic jet velocities and a dispersion process that is momentum-dominated. Under these conditions, the Gaussian dispersion model unrealistically overestimates the amount of hydrogen in the explosive region and the distance to the lower flammable region. Therefore, these properties of gaseous releases were calculated using a jet dispersion model described in reference (2).

The results of this modeling are shown in figure 4-3 as minimum separation distances versus inside diameter of the pipe. The upper curve is the maximum distance to the lower flammability limit of 4% hydrogen. Each utility shall determine that the location of air pathways into safety-related structures exceeds this minimum separation distance or show that other criteria should be applied to a specific case. An example of such an exception would be if the air intakes have automatic shutters controlled by hydrogen analyzers thus preventing the ingestion of a flammable mixture.

The lower curve on figure 4-3 is the minimum required distance to safety-related structures with greater than or equal to an 8-inch thick reinforced mansonry or concrete wall. This distance includes the drift distance of an unignited, fully developed gaseous jet plus the blast distance for the maximum amount of hydrogen in the detonable region. It conservatively assumes that the pipe break is oriented directly toward the safety-related structures. Each utility shall determine compliance with this minimum separation distance or demonstrate that other criteria should be applied.

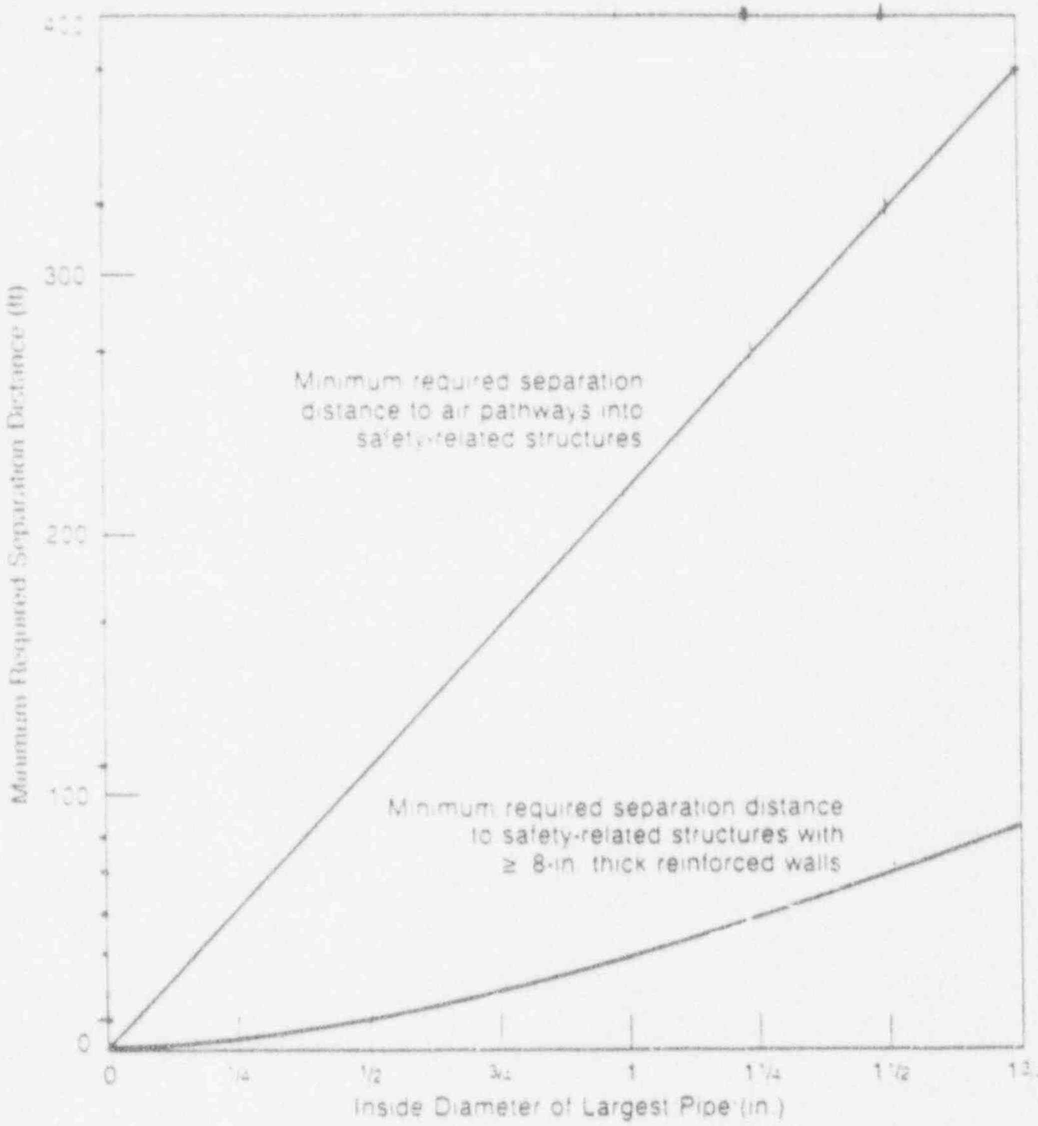

Figure 4-3 Minimum required separation distance versus ID of pipe for releases from 2450 psig gaseous hydrogen storage systems

4.2 LIQUID HYDROGEN

4.2.1 Storage Vessel Failure

For this report, storage vessel failure is defined as a large breach resulting in the rapid emptying of the entire contents of liquid hydrogen. It is assumed that the tank is full at the time of failure and that the entire spill vaporizes instantaneously. The following enumerates potential causes of vessel failure and the required design features that mitigate or alleviate these potentials.

- Seismic

  The tank and its foundation shall be designed to meet the seismic criterion for critical structures and equipment at the plant site (i.e., design basis earthquake). It is preferable to seismically support all liquid hydrogen piping. If this is not possible, the liquid hydrogen piping shall be seismically supported up to and including excess flow protection devices. The specific liquid hydrogen tank and piping design at each installation shall meet these requirements.

- Tornado and Tornado Missiles

  The tank and its foundation shall be designed to withstand the "design basis tornado characteristics" as outlined in Regulatory Guide 1.76. As a minimum, the tank shall remain in place so that any liquid spillage will originate from the tank location. The specific tank and foundation design at each installation shall meet these requirements.

  Design basis tornado-generated missiles are capable of breaching all known commercially available liquid hydrogen storage vessels. Therefore, tornado missiles are a potential cause of "storage vessel failure."

- Aircraft

  A large aircraft crashing directly into the storage area is capable of breaching all known commercially available liquid hydrogen storage vessels. Therefore, aircraft crash is a potential cause of "storage vessel failure."

- Fire

  The overpressure protection system shall be sized to accommodate the worst-case vaporization rate caused by a hydrocarbon fire engulfing the outer shell with loss of vacuum and hydrogen in the annulus of the double-wall storage tank (as per Compressed Gas Association 5.3 and ASME Section VIII requirements).

- Flood

  The following flood conditions could result in vessel failure:

  --High water reaches the top of the vent stack for the overpressure protection system.

  --High flood velocities dislodge the tank.

  Under either condition, water could enter the vent system and defeat the overpressure protection system. Therefore, the tank shall be located such that maximum flood heights cannot exceed the vent stack elevation and such that potential flood velocities cannot damage the vent stack or dislodge the tank.

- Vehicle Impact

  The storage vessel shall be protected from the impact of the largest vehicle used on-site by a barricade capable of stopping such a vehicle.

- Vessel Structural Failure

  The storage vessel shall be designed, constructed, inspected and operated to assure an extremely low likelihood of tank structural failure during its tenure on site. A vessel designed in accordance with this document complies with this low-probability requirement.

4.2.1.1 Fireball. For the two potential causes of "storage vessel failure," tornado missiles and aircraft impact, a fireball at the tank location is the expected result. The major reasons for this is the high ignitability of hydrogen and the density of ignition sources in the aftermath of these causal events. An aircraft impact or a design basis tornado and the associated missles will also provide numerous sources of ignition from downed power lines, damaged transformers, and switchgears, etc. Details of these considerations are given in the report for the Dresden plant (2).

The thermal flux versus distance from the fireball center (tank location) is shown on figure 4-4 for the range of commercially available tank sizes. The durations of the various fireball sizes are also given. These fluxes and durations will not adversely affect equipment or personnel enclosed in concrete/steel safety-related structures. However, each utility shall review any unique site characteristics to assure all safety-related equipment will function in the event of a fireball.

4.2.1.2 Explosion at Tank Site. Although an explosion is not expected, safety-related structures and equipment shall be verified to be capable of withstanding a detonation occurring at the site of the tank installation. For the instantaneous

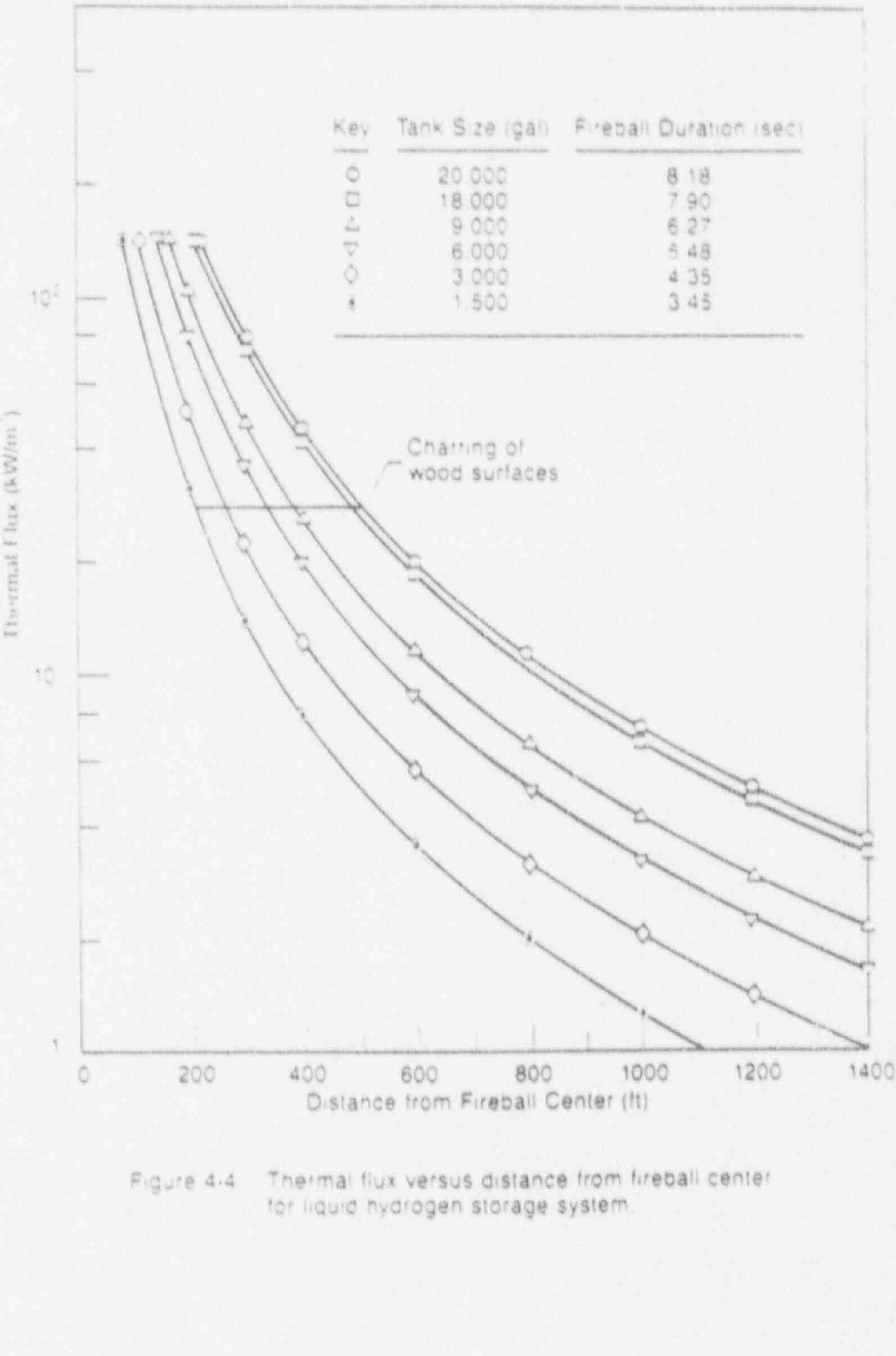

Figure 4-4 Thermal flux versus distance from fireball center for liquid hydrogen storage system.

release of the entire tank contents, the following were used to determine blast parameters for an explosion at the tank site:

1. Gaussian F weather stability
2. Detonation limits of hydrogen, 18.3-59%
3. TNT - hydrogen equivalent of 20% on an energy basis (520% on a mass basis)

NUREG/CR-2726 reports that detonations have been observed for hydrogen concentrations as low as 13.8% when ignited in a long, large-diameter tube. The explosive yield or TNT equivalence of such threshold concentration reactions is extremely low because most of the combustion energy is expended in the transition to detonation. This is essentially the reason why it represents the lower detonation limit; any less concentration will give a zero detonation yield. This also points out that both hydrogen concentration and explosive yield affect the total equivalent mass of TNT for a given release.

Regulatory Guide 1.91 models the blast effects from transportation accidents by assuming 100% of the cargo detonates at a TNT mass equivalence of 240% (one pound of cargo equals 2.4 pounds of TNT). The analysis described in this report modeled large spills of hydrogen by calculating the amount of release that is between 18.3 and 59% (~46% of the vessel contents) and assuming that it detonates at a TNT mass equivalence of 520%. The resulting TNT equivalence for this method is one pound of vessel contents equals 2.4 pounds of TNT, an identical result to that obtained with the NRC method.

The above results in an equivalence of 1.37 lbs of TNT per gallon of tank size. Using this conversion factor and U.S. Army Technical Manual TM5-1300 and the damage criteria outlined in appendix B, required separation distances have been determined as a function of tank size. The results are shown on figure 4-5 for the design parameters of the three building types described in section 4.1.2.2. For buildings with other design parameters, the methods in appendix B or in NUREG/CR-2462 (1) may be used to determine separation distances. Each utility shall use these methods for determining the minimum required separation distances from the storage tank to safety-related structures or equipment for the event of an explosion at the tank site.

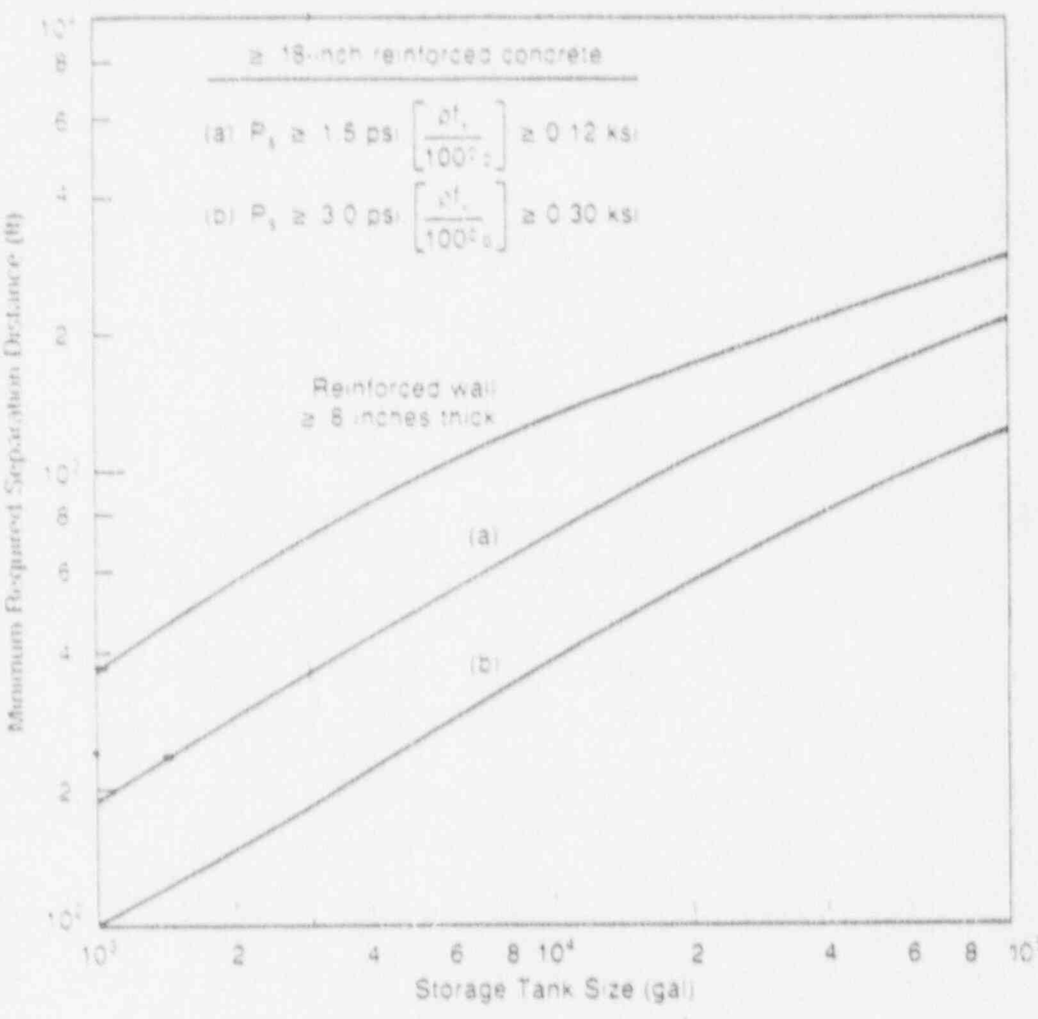

Figure 4-5 Minimum required separation distance versus liquid hydrogen storage tank size for instantaneous release of entire tank contents and explosion at tank site F weather stability

4.2.2. Pipe Breaks

This section addresses the requirements for gaseous and liquid hydrogen piping systems attached to the storage vessel up to the point where excess flow protection is provided. The criteria for acceptable siting for the event of a pipe break are the same as outlined in section 4.1.3. It is conservatively assumed that all releases occur while the storage vessel is at 150 psig (the maximum allowable working pressure of the majority of commercially available tanks).

4.2.2.1. Gaseous Piping. The same dispersion model for momentum-dominated jets discussed in section 4.1.3 applies to gaseous releases from liquid storage tank piping with the appropriate release conditions for saturated vapors. The results of this modeling are shown in figure 4-6 as minimum separation distances versus hole size or inside diameter of piping not protected with excess flow devices. The upper curve is the maximum drift distance to the lower flammability limit and is the minimum required separation distance to air pathways into safety-related structures. The three lower curves are required separation distances for the representative types of safety-related structures. These distances are the sum of both the drift and blast distances. Structures with other parameters can be analyzed using the methods in appendix B or in NUREG/CR-2462 (1). Each utility shall determine that the storage vessel piping and location meet these minimum requirements or show that less stringent criteria should be applied to a specific case. An example of such a suitable exception would be if the air intakes are provided with automatic shutters controlled by hydrogen analyzers to prevent the ingestion of a flammable mixture.

4.2.2.2 Liquid Piping. The vapor cloud formed by the flashing and rapid vaporization of a liquid release is nearly neutrally buoyant and has little momentum associated with its formation. For these conditions, a Gaussian dispersion model is employed using the following conservative assumptions:

1. Instantaneous vaporization of release
2. F weather stability
3. 1 m/s wind speed
4. Wind direction towards safety-related area

No credit is to be taken for site-specific wind direction or speed characteristics since it is assumed that pipe breaks can occur during the worst-case weather and wind conditions.

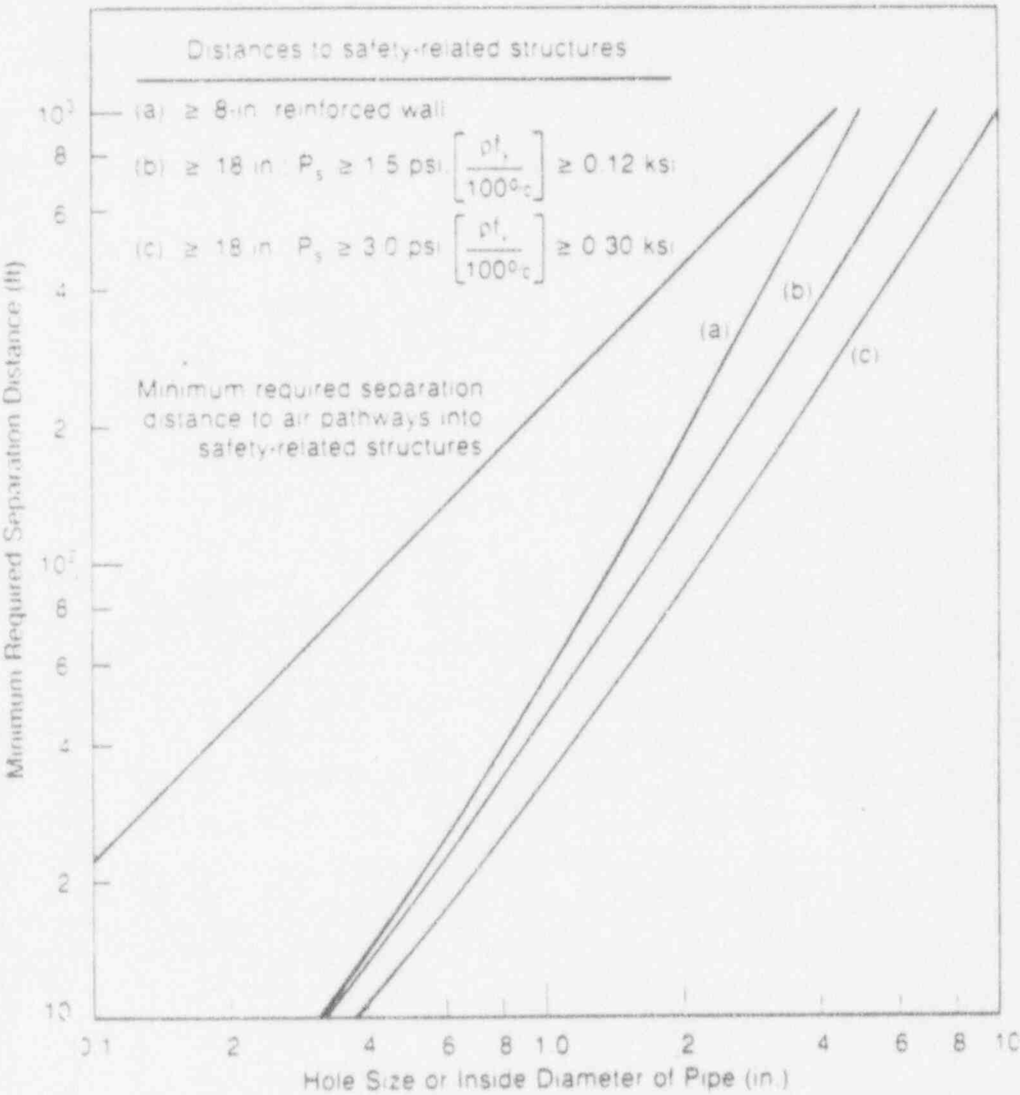

Figure 4-6 Minimum required separation distance versus hole size and ID of pipe for gaseous releases from 150 psig liquid hydrogen storage tank

The minimum required separation distances for liquid hydrogen pipe breaks, using the above assumptions, are given on figure 4-7 as a function of discharge rate and hole size. The upper curve is the drift distance to the lower flammability limit for a fully developed could with F stability and 1 m/s windspeed. This defines the minimum required separation distance to air pathways into safety-related structures. The three lower curves define the minimum required separation distances to the representative safety-related structures. These curves include the drift distance to the center of the detonable cloud and the blast distance for the amount of hydrogen in the detonable region. For other structure types, appendix B or NUREG/CR-2462 (1) may be used to determine blast distances. These distances shall be applied to all liquid piping, including those from any pump discharges, that are not seismically supported or protected by excess flow devices.

4.3 ELECTROLYTIC

4.3.1 General

The electrolytic supply option need not constitute storage of hazardous materials on-site if it operates at approximately atmospheric pressure and involves the storage of no more than 2500 scf of hydrogen and 250 scf of oxygen. If these limits are met, and the system is designed as described in section 3.3, it need only be analyzed as described below. Other system designs have not yet been considered. Compressed gases utilized in conjunction with electrolytic systems shall be in accordance with sections 3.1 and 4.1.

Events important to industrial safety (abnormal transients, accidents and external events) must be evaluated to identify those which could result in any of the following conditions:

1. Hydrogen accumulation to a combustible mixture in an enclosed space.
2. Air or oxygen mixing with hydrogen within electrolytic system components.
3. Hydrogen fires.

When the potential exists for the above undesired conditions to occur, appropriate mitigating features shall be incorporated in the design or operation of the system or the consequences with respect to plant and personnel safety shall be evaluated by the owner.

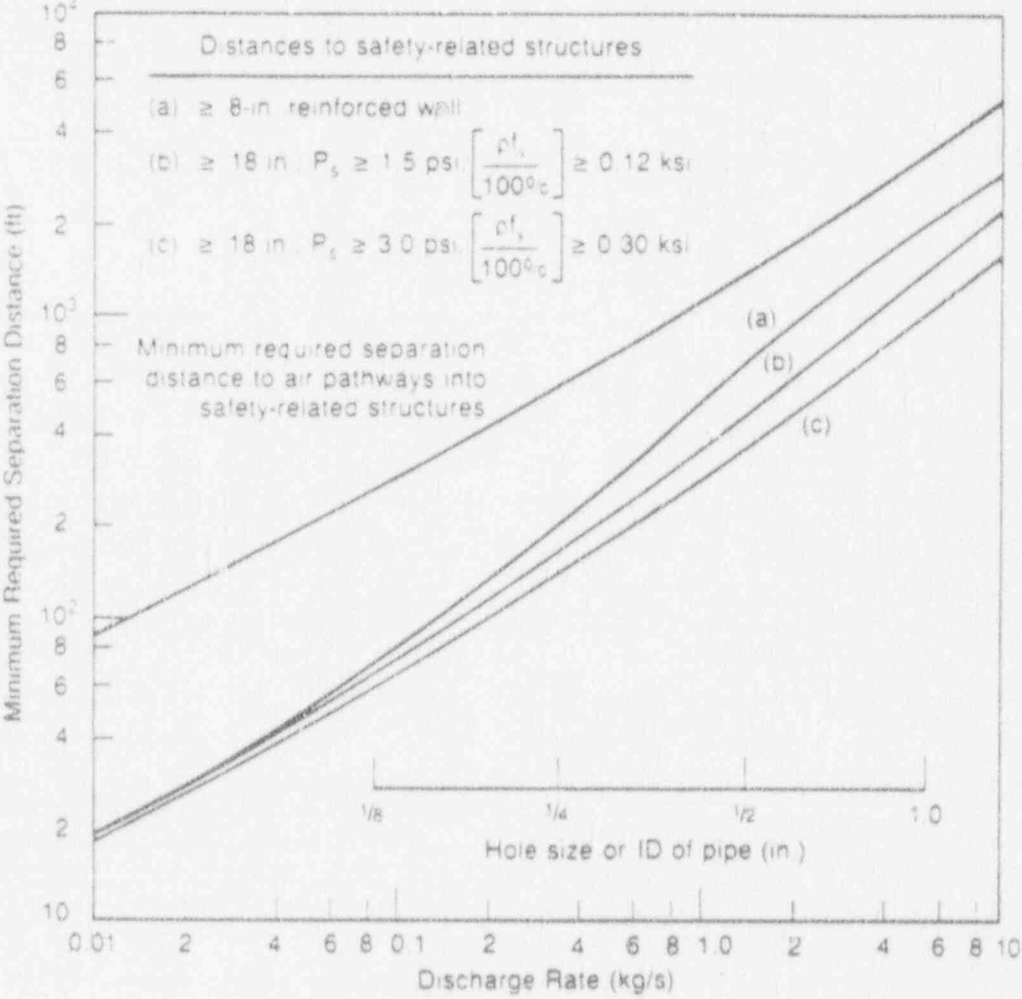

Figure 4-7. Minimum required separation distance versus hole size and discharge rate for liquid releases from 150 psig liquid hydrogen storage tank F weather stability, 1 m/s wind velocity

4.3.2 Purity of Gases

The gases as collected from the electrolytic cells in a well working system will be over 99% pure and concentration of the oxygen in the hydrogen stream and the concentration of hydrogen in the oxygen stream will be well below the ignition limits. However, over time, purity may tend toward unacceptable limits due to the buildup of oxides or contaminants on the electrodes. This trend is a very slow process detectable by periodic purity testing well before combustible mixtures are reached. The time that it takes depends on materials of the electrodes, and impurities in the water. To monitor cell performance and avoid combustible mixtures, gas purity shall be periodically or continuously measured. As a second precaution against an unsafe condition, the equipment shall be designed to contain an internal explosion.

4.3.3 Air Inleakage

The electrolytic cells and their gas collection headers shall be controlled to a pressure above atmospheric.

Since nearly any method of compression will cause a reduction of pressure at the inlet of the compression device, the equipment at and between the pressure regulating device (for maintaining the gas generator pressure) and the compressor must be designed to avoid air inleakage. This equipment shall be designed to (1) not contain sufficient hydrogen to represent a hazard to plant safety, (2) not have any ignition sources in the hydrogen flow path, and (3) avoid combustible gas mixtures. Valves in this flow path should have spark-resistant seat and stem guides. The design should be capable of containing an internal explosion.

4.3.4 Out Leakage

The system must be designed to avoid combustible gas mixtures which could result from unintentional outleakage. Controlled venting to safe locations in the atmosphere is acceptable.

The kindling temperatures of combustible materials decrease with increased concentrations of oxygen. Therefore, oxygen must not be vented in the vicinty of combustible materials that would be at temperatures above the kindling temperature in a pure oxygen concentration.

4.3.5 External Events

External events such as seismic, tornado, aircraft crash and flood cannot result in consequences more severe than cited above and need not be considered further.

4.4 LIQUID OXYGEN

4.4.1 Site Characteristics of Liquid Oxygen

4.4.1.1 Overview. Review of the following site characteristics shall be completed by each BWR facility as part of their efforts to locate the liquid oxygen storage system.

1. Location of supply system in proximity to exposure as addressed in NFPA 50.
2. Route of liquid oxygen delivery on site.
3. Location of supply system in proximity to safety-related equipment.
4. Location of hydrogen storage.

4.4.1.2 Specific Considerations.

4.4.1.2.1 Fire protection. The area selected for liquid oxygen system siting shall meet or exceed all requirements for protection of personnel and equipment as addressed in NFPA 50, Bulk Oxygen Systems. The standard identifies the types of exposures under consideration. The number of exposures warrants a plant-specific review for proper code compliance. As much separation distance as practical should be provided between the hydrogen and oxygen systems.

4.4.1.2.2 Security. All liquid oxygen supply system installations shall be completely fenced, even when located within the security area. Lighting shall be installed to facilitate night surveillance.

4.4.1.2.3 Route of liquid oxygen delivery on site. Each plant should determine the route to be taken by liquid oxygen delivery trucks through on- and off-site areas. In order to protect the oxygen storage area from any vehicular accidents, truck barriers shall be installed around the perimeter of the system installation.

within the plant security area all deliveries shall be controlled by plant security personnel, per the requirements of 10 CFR 73.55.

4.4.1.2.4 Location of storage system to safety-related equipment. Each plant shall determine that the location of the liquid oxygen supply system is acceptable considering the hazard described in sections 4.4.2 and 4.4.3.

4.4.2 Liquid Oxygen Storage Vessel Failure

Liquid oxygen storage vessels are vulnerable to the same potential causes of failure as the liquid hydrogen vessels but the potential consequences of failure are much less severe. The potential threat from a liquid oxygen spill is the contact of oxygen-enriched air with combustible materials or the ingestion of oxygen-enriched air into safety-related air intakes. Additional information on the effects of oxygen-enriched atmospheres is given in NFPA 53M and in ASTM G63-83a and G88-84. Fc ne purpose of this report, it is conservatively assumed that total oxygen concentrations above 30 vol% (21% O_2 in air + 9% enriched O_2) will increase the effective combustibility of ignitible materials in the area.

4.4.3 Liquid Oxygen Vapor Cloud Dispersion

The vapor cloud instantaneously formed by a large liquid oxygen spill will have a density of 3.59 relative to air. Such a cloud will experience considerable gravity-driven slumping as it disperses and translates with the wind. This process has been described by the DEGADIS model developed by Prof. J. A. Havens of the University of Arkansas (3). His model has been found to agree well with published data on large releases of dense gases conducted by the U.S. Department of Energy, U.S. Coast Guard and others.

The DEGADIS model has been used to determine the height of the vapor cloud as a function of distance for various sizes of commercially available liquid oxygen storage tanks. It was conservatively assumed that any vessel failure would result in the instantaneous vaporization of the entire tank contents. The curves on figure 4-8, which define "acceptable location of safety-related air intake," were generated by using the DEGADIS model under the worst-case weather conditions of F stability and 10 m/s wind speed for total oxygen concentrations of 30 vol%. For dense gas dispersion, lower wind speeds result in more radial spreading with a lower cloud height and shorter maximum drift distance. Higher wind speeds will translate even the largest release past safety-related intakes in less than 10 sec, giving little time for ingestion of enriched air.

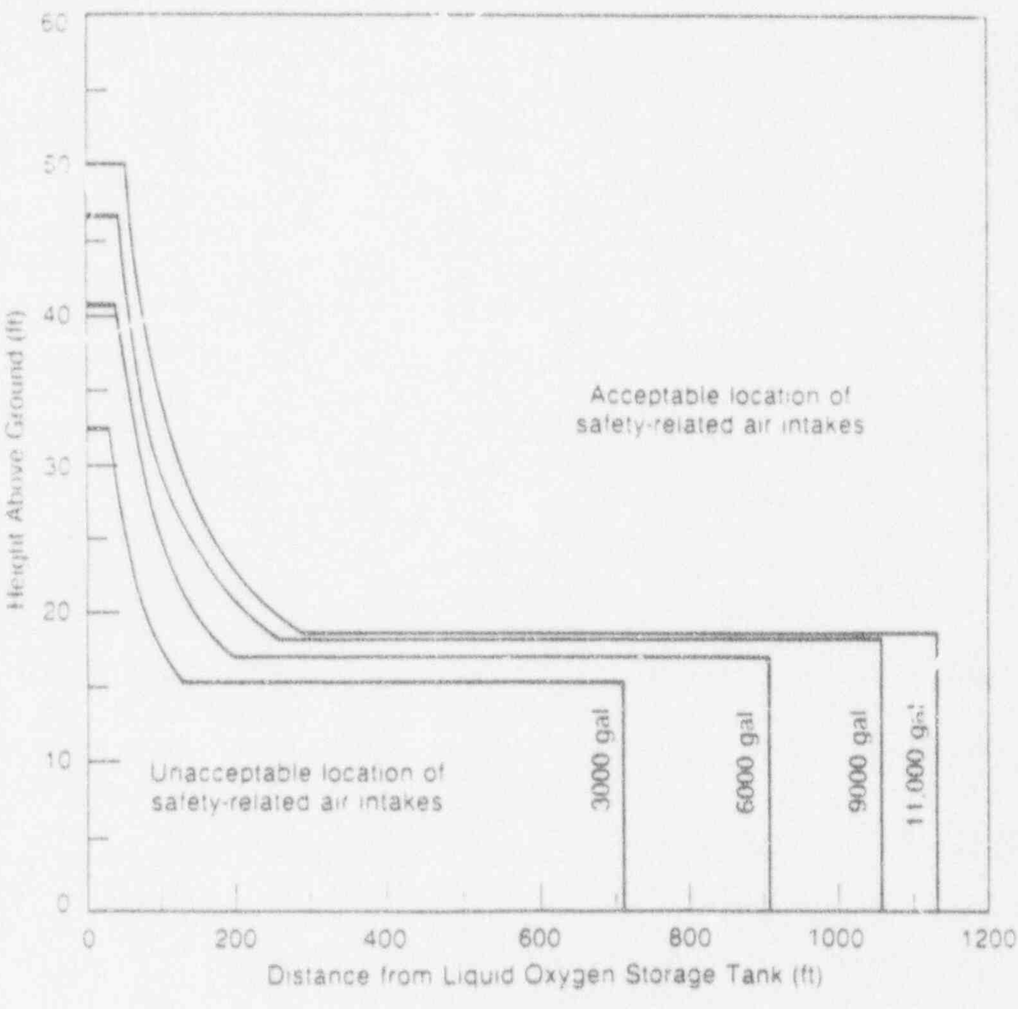

Figure 4-8 Acceptable locations of safety-related air intakes for various sizes of liquid oxygen storage tanks.

Therefore, liquid oxygen storage vessels shall be located such that safety-related air intakes are within the acceptable region defined by figure 4-8 or alternative analyses shall be performed to justify the location. Since this figure assumes the origin of release is from the storage location, the tank and its foundation shall be designed to remain in place for both design basis tornadoes and site-specific flood conditions.

4.5 REFERENCE

1. R. P. Kennedy, T. E. Blejwas, and D. E. Bennett. "Capacity of Nuclear Power Plant Structures to Resist Blast Loadings." NUREG/CR-2462. Sandia National Laboratories for U.S. Nuclear Regulatory Commission.

2. "Air Products Liquid Hydrogen Storage System Hazardous Consequence Analysis." Revision 1, October 1, 1985.

3. J. A. Havens. "The Atmospheric Dispersion of Heavy Gases: An Update." IChemE Symposium Series No. 93, 1985.

Section 5

VERIFICATION

The various methods of verifying the effectiveness of HWC (i.e., electrochemical potential, constant extension rate tests, etc.) are not within the scope of this document. Appropriate methods of verification should be selected and implemented on a plant-specific basis.

Section 6

OPERATION, MAINTENANCE, AND TRAINING

This section gives recommendations to the operating utility for operation, maintenance, and training in order to meet the design intent of the hydrogen water chemistry (HWC) system.

The operation of a HWC system will require operator and chemistry personnel attention. Because of the radiation increases that result from employing this system, an awareness of ALARA principles is required by all plant personnel. This system could also have an effect on the off-gas system and the plant fire protection program.

6.1 OPERATING PROCEDURES

Written procedures describing proper valving alignment and sequence for any anticipated operation should be provided for each major component and system process. Check-off lists should be developed and used for complex or infrequent modes of operation. Operating procedures should be considered for the following operations:

1. Hydrogen addition system startup, normal operation, shutdown and alarm response.
2. Material (gas or liquid) handling (filling of storage tanks) operations that are consistent with the supplier's recommendations.
3. Purging of hydrogen and oxygen lines.
4. Operation of on-site gas generation system (if appropriate).
5. Fire protection or safety measures for hydrogen- or oxygen-enhanced fires and hydrogen or oxygen spills.
6. Calibration and maintenance procedures as recommended by equipment or gas suppliers.
7. Routine inspection of HWC system equipment.
8. Adjustment of the main steam line radiation monitor setpoints (if appropriate).

6.1.1 Integration Into Existing Plant Operation Procedures

Where appropriate, operation of the HWC system shall be incorporated into normal plant procedures such as plant startup and shutdown.

6.1.2 Plant-Specific Procedures

Appropriate procedures shall be developed to provide guidance for plant operators when operation of the HWC system necessitates operation of an existing system in a different mode or raises new concerns. Areas which should be considered are:

1. Operation of the off-gas system
2. Possible off-gas fires

6.1.3 Radiation Protection Program

Operation of an HWC system results in an increase in radiation levels wherever nuclear steam is present. The radiation protection program shall be reviewed and appropriate changes made to compensate for these increased radiation levels.

The following guidelines are established to ensure that radiological exposures to both plant personnel and the general public are consistent with ALARA requirements. Compliance with these requirements minimizes radiologically significant hazards associated with HWC implementation. The operation of a hydrogen addition system may cause a slight reduction in the off-gas delay time due to the increase in the flow rate of noncondensables resulting from the excess oxygen added. This may slightly increase plant effluents and should be reviewed on a plant-specific basis.

6.1.3.1 ALARA Commitment. Permanent hydrogen water chemistry systems and programs will be designed, installed, operated, and maintained in accordance with the provisions of Regulatory Guides 8.8 and 8.10 to assure that occupational radiation exposures and doses to the general public will be "as low as reasonably achievable."

6.1.3.2 Initial Radiological Survey. A comprehensive radiological survey should be performed with hydrogen injection to quantify the impact of hydrogen water chemistry on the environs dose rates, both within and outside the plant. This survey should be used to determine if significant radiation changes occur within the plant and at the site boundary. Based upon the magnitude of the change, it should be determined if new radiation areas or high radiation areas need to be created. Appropriate posting, access, and monitoring requirements should be implemented for the affected areas. Plant operating and surveillance procedures should be revised, as required, to minimize the time and number of personnel required in radiation areas for operations, maintenance, in-service inspection, etc.

6.1.3.3 <u>Plant Shielding.</u> The radiological survey of subsection 6.1.3.2 should be used to determine the adequacy of existing plant shielding. In addition, the radiation levels from sample lines, sample coolers and monitoring equipment may increase due to HWC and should be checked for adequate shielding. If required, measures for selective upgrading of plant shielding should be implemented to reduce both work area and site boundary dose rates.

6.1.3.4 <u>Maintenance Activities.</u> Hydrogen water chemistry will have minimum impact on occupational exposures resulting from maintenance activities. Plant procedures should incorporate appropriate requirements for access to and monitoring of areas where increased dose rates exist with HWC to satisfy ALARA requirements. For extended maintenance, plant procedures should include provisions to terminate the hydrogen injection. Due to the short half-life of N-16, radiation levels will return to pre-HWC conditions within minutes of hydrogen shutoff.

6.1.3.5 <u>Radiological Surveillance Programs.</u> Dose rate surveys should be conducted and radiation levels should be monitored periodically to ensure compliance with the radiological limits imposed by 40 CFR Part 190, 10 CFR Part 100, and 10 CFR Part 20. Additional surveys may be required to comply with ALARA requirements. Hydrogen water chemistry, in association with improved water quality operational practices, could affect the crud buildup within the recirculation piping and the shutdown dose rates. A radiological surveillance program should be established to monitor shutdown dose rates and crud buildup over a number of fuel cycles to evaluate possible changes.

6.1.3.6 <u>Measurement of N-16 Radiation.</u> The radiological surveillance program should include provisions for the new distribution of N-16 in the main steam. Selection of appropriate health physics instrumentation and application of correction factors are required to provide accurate dose measurements. (This correction is required due to the effect of the energetic N-16 gamma on instrumentation calibrated with less energetic gamma sources.) All plant survey meters should be reviewed and appropriate calibration and correction methods accounted for in plant procedures.

A review of the plant personnel dosimetry program shall be conducted to ensure that the appropriate calibration or correction factors are used.

6.1.3.7 Value/Impact Considerations. The following discussion reviews the total dose impact on a plant which implements HWC.

A radiological assessment at Dresden indicates that the total dose increase with HWC is approximately 0.5% on an annual basis (from 1935 to 1945 man-rem/year) (1). While this increase is site dependent due to plant layout and shielding configurations, significant variances from the Dresden assessment are not anticipated. Thus, over the life of a plant (assuming a 25-year remaining life), the projected total dose increase with HWC is ~250-300 man-rems.

With HWC implementation, the potential exists to relax current augmented in-service inspection requirements imposed by NRC Generic Letter 84-11 (2) and elimination of extended plant outages for pipe replacement and/or repair. The value/impact assessment presented in appendix E to reference 3 projects a 1161 man-rem (best estimate) savings over the life of the plant as a consequence of reduced inspections and repairs with HWC. Typical pipe replacement projects result in a total dose of 1400 to 2000 man-rem. Thus, HWC implementation could result in a significant savings in total dose over the life of the plant.

6.1.4 Water Chemistry Control

Procedures should be developed to maintain the high reactor water quality necessary to obtain the maximum benefit from the HWC system. Intergranular stress corrosion cracking can be mitigated by controlling the ionic impurity content of the primary coolant and by reducing the dissolved oxygen level in the primary coolant by use of HWC. The EPRI-BWR Owners Group have developed "BWR Hydrogen Water Chemistry Guidelines" (4), which must be met in order to obtain the full benefits of HWC. These water chemistry guidelines should be used as a basis for developing a plant-specific water chemistry control program.

Hydrogen water chemistry can reduce the dissolved oxygen level in the condensate and feedwater. It has been shown that at very low levels of dissolved oxygen, corrosion and metal transport to the primary system would be increased. If, when operating on HWC, the dissolved oxygen concentration drops below 20 ppb, an evaluation should be made to determine if there is increased corrosion or metals tranport, or if other factors relating to such a reduced oxygen concentration need to be considered. If this evaluation determines that oxygen injection is necessary, a system should be designed using the guidance provided in sections 2.3.2 and 3.4 of this report.

6.1.5 Fuel Surveillance Program

No significant effect of hydrogen injection on fuel performance has been observed, nor is expected. However, since in-reactor experience with hydrogen water chemistry is limited, utilities should consider the fuel surveillance programs recommended by their fuel suppliers.

6.2 MAINTENANCE

A preventative maintenance program should be developed and instituted to ensure proper equipment performance to reduce unscheduled repairs. All maintenance activities should be carefully planned to reduce interference with station operation, assure industrial safety, and minimize maintenance personnel exposure. Written procedures should be developed and followed in the performance of maintenance work. They should be written with the objective of protecting plant personnel from physical harm and radiation exposure, and of reducing hydrogen addition system downtime. Radiation exposure should be reduced by shortening the time required in a high radiation field and by reducing its intensity by turning off the HWC system or other means during the maintenance period.

All excess flow check valves used for hydrogen line break protection shall be periodically tested to assure they will function properly.

6.3 TRAINING

In order for the HWC system to maintain its system integrity and to provide the expected benefits from its use, the system must be operated correctly. The most effective means of reducing the potential of operator error is through proper training.

Training should be provided to:

1. Instruct operators on the function, theory and operating characteristics of the system and all its major system components.

2. Advise operators of the consequences of component malfunctions and misoperation and provide instruction as to appropriate corrective actions to be taken.

3. Advise operations and maintenance personnel of the potential hazards of gases in the system, and provide instruction as to appropriate procedures for their handling.

4. Instruct emergency response personnel on appropriate procedures for handling fires or personnel injuries involving spills or releases of H_2 or O_2 liquid and gases.

5. Instruct plant personnel on the expected radiation changes due to the operation of the HWC system and the appropriate ALARA practices to be taken to minimize dose.

6. Instruct appropriate personnel on the benefits of HWC.

7. Advise maintenance and construction personnel of the routing of hydrogen lines and of the appropriate protective actions to be taken when working near these lines.

Periodic training should be provided to reinforce information described above and to communicate information regarding any modifications, procedural changes, or incidents.

6.4 IDENTIFICATION

In order to aid plant personnel in identifying hydrogen and oxygen lines, these lines should be color coded as required by ANSI A13.1.

6.5 REFERENCES

1. "Environmental Impact of Hydrogen Water Chemistry." EPRI Hydrogen Water Chemistry Workshop, Atlanta, Georgia, December 1984.

2. "Inspection of BWR Stainless Steel Piping." NRC Generic Letter 84-11, April 19, 1984.

3. "Report of the United States Nuclear Regulatory Commission Piping Review Committee." NUREG-1061, Volume 1, August 1984.

4. BWR Hydrogen Water Chemistry Guidelines: 1987 Revision. NP-4947-SR-LD. Palo Alto, Calif.: Electric Power Research Institute, to be published.

Section 7

SURVEILLANCE AND TESTING

7.1 SYSTEM INTEGRITY TESTING

In addition to the testing required by the applicable design codes, completed process systems which will contain hydrogen shall be leak tested with helium or a soap solution as appropriate prior to initial operation of the system. All components and joints shall be so tested in the fabrication shop or after installation, as appropriate. Appropriate helium leak tests shall be performed on portions of the system following any modifications or maintenance activity which could affect the pressure boundary of the system.

7.2 PREOPERATIONAL AND PERIODIC TESTING

Completed systems should be tested to the extent practicable to verify the operability and functional performance of the system. Proper functioning of the following items should be verified:

1. Trip and alarm functions per table 2-2.
2. Gas purity, if generated on site.
3. Safety features.
4. Excess flow check valves.
5. System controls and monitors per table 2-2.

A program should be developed for periodic retesting to verify the operability and the functional performance of the system.

Section 8

RADIATION MONITORING

8.1 INTRODUCTION

This section reviews the radiological consequence of hydrogen water chemistry (HWC) and presents the basis for increasing the main steam line radiation monitor set point to accommodate HWC. It is concluded that implementation of HWC does not reduce the margin of safety as defined in the basis of the technical specification set point.

During normal operation of a BWR, nitrogen-16 is formed from an oxygen-16 (N-P) reaction. N-16 decays with a half-life of 7.1 sec and emits a high-energy gamma photon (6.1 MeV). Normally, most of the N-16 combines rapidly with oxygen to form water-soluble, nonvolatile nitrates and nitrites. However, because of the lower oxidizing potential present in a hydrogen water chemistry environment, a higher percentage of the N-16 is converted to more volatile species. As a consequence, the steam activity during hydrogen addition can increase up to a factor of approximately five. The dose rates in the turbine building, plant environs, and off-site also increase; however, the magnitude of the increase at any given location depends upon the contribution of the steam activity to the total dose rate at that location. The specific concerns include:

1. The dose to members of the general public (40 CFR 190),
2. The dose to personnel in unrestricted areas (10 CFR 20), and
3. The maintenance of personnel exposure "as low as reasonably achievable" (ALARA).

8.2 MAIN STEAM LINE RADIATION MONITORING

As noted in the previous section, main steam line radiation levels can increase up to approximately fivefold with hydrogen water chemistry. The majority of BWRs have a technical specification requirement for the main steam line radiation monitor (MSLRM) set point that is less than or equal to three (3) times the normal rated full-power background. For these plants an adjustment in the MSLRM set point may be required to allow operation with hydrogen injection. For earlier

BWRs with MSLRM set points of seven (7) to ten (10) times normal full-power background, a set point change may not be required.

8.2.1 Dual MSLRM Set Point Recommendation

For plants at which credit is taken for an MSLRM-initiated isolation in the control rod drop accident (CRDA), a dual set point approach may be utilized. At most plants, the MSLRM set point is specified in the plant Technical Specifications (Tech Specs) as some factor times rated full-power radiation background. With hydrogen addition, the full-power background could increase up to 5 times that without hydrogen addition. Below 20% rated power or the power level required by the FSAR or Tech Specs (see table 2-1), the existing set point is maintained at the Tech Spec factor above normal full-power background, and hydrogen should not be injected. About 20% rated power, the MSLRM set point should be readjusted to the same Tech Spec factor above the rated full-power background with hydrogen addition. This adjustment would be made by the plant personnel during startups and shutdowns. Plant power would remain constant during this adjustment process. Thus, the Tech Spec factor by which the MSLRM sec point is adjusted remains the same with and without hydrogen addition, but the background radiation level increases with hydrogen addition. If an unanticipated power reduction event occurs such that the reactor power is below this power level without the required set point change, control rod motion should be suspended until the necessary set point adjustment is made. At newer plants, credit is not taken for an MSLRM-initiated isolation after a CRDA, and a dual set point is not needed at these plants.

Plants that need a dual set point should consider changing their Technical Specifications to increase the factor used to determine the MSLRM set point, if their CRDA analysis will permit this increase. A suggested approach would be to use the Susquehanna Steam Electric Station, unit 1, amendment no. 58 Technical Specification change as a model. Under this approach, the MSLRM set point was raised based on a satisfactory evaluation of the off-site consequences.

8.2.2 MSLRM Safety Design Basis

The only design basis event for which some plants may take credit for main steam isolation valve (MSIV) closure on main steam line high radiation is the design basis control rod drop accident (CRDA). As documented in reference (1), the CRDA is only of concern below 10% of rated power. Above this power level the rod worths and resultant CRDA peak fuel enthalpies are not limiting due to core voids and faster Doppler feedback. Since the current MSLRM set point will not be changed below 20% rated power, the MSLRM sensitivity to fuel failure is not impacted and the FSAR analysis for the CRDA remains valid.

The licensing basis for the CRDA states that the maximum control rod worth is established by assuming the worst single inadvertent operator error (2). From references (2) and (3), the maximum control rod worth above 20% rated power, assuming a single operator error, is <0.8% $\Delta K/K$. Parametric studies utilizing the conservative GE excursion model (1) indicate that the maximum peak fuel enthalpy for a dropped control rod worth of 0.8% $\Delta K/K$ is less than 120 calories per gram (3). Consequently, the conservatively calculated peak fuel enthalpy for a CRDA above 20% rated power will have significant margin to the fuel cladding failure threshold of 170 calories per gram.

An increase in the MSLRM set point will not impact any other FSAR design basis accident or transient analysis since no credit is taken for this isolation signal. Consequently, a technical specification change which adopts the recommended dual set point approach will not reduce overall plant safety margins.

8.2.3 MSLRM Sensitivity

Conceptually, the sensitivity of the MSLRM to fission products is effectively reduced by the increase in the set point above 20% power. However, it is still functional and capable of initiating a reactor scram. The main function of the instrument is to help maintain off-site releases to within the applicable regulatory limits. The MSLRM is supplemented by the off-gas radiation monitoring system which monitors the gaseous effluent prior to its discharge to the environs. The off-gas radiation monitor set point is established to help ensure that the equivalent stack release limit is not exceeded.

8.2.4 Conclusions

From the above discussion, it can be concluded that an increase in the MSLRM set point above 20% rated power will not reduce the safety margins as defined by Technical Specifications or increase the off-site radiological effects as a consequence of design base accidents. Furthermore, since this change to the MSLRM can be justified independent of HWC, this change does not constitute an unreviewed safety concern.

8.3 EQUIPMENT QUALIFICATION

Outside primary containment the increase in dose rates with HWC is relatively small relative to the integrated dose assumed for equipment qualification (EQ) tests. Furthermore, dose rates inside the drywell near the recirculation piping will decrease because of the increased carry-over of N-16 in the steam. Each utility should review the resultant dose increases to ensure that the doses assumed in the EQ tests required for electrical equipment per 10 CFR Part 50.49 remain bounding.

8.4 ENVIRONMENTAL CONSIDERATIONS

Implementation of an HWC system is unlikely to significantly increase the amounts or significantly change the types of effluents that may be released off-site. Although an increase in individual or cumulative occupational radiation exposure may occur, the guidelines provided in section 6.1.3 of this document will ensure that radiological exposures to both plant personnel and the general public are consistent with ALARA requirements. Since the design objectives and limiting conditions for operation as defind by 10 CFR Part 50, Appendix I, are not impacted, no Appendix I revision is required.

Each plant should examine the environmental effects of an HWC system. However, it is unlikely that environmental impact statements or environmental assessments will be required for HWC systems.

8.5 REFERENCES

1. R. C. Stirn et al. "Rod Drop Analysis for Large Boiling Water Reactors." NEDO-10527. General Electric Company, March 1972.

2. R. C. Stirn et al. "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores." NEDO-10527, Supplement 2. General Electric Company, January 1973.

3. R. C. Stirn et al. "Rod Drop Accident analysis for Large Boiling Water Reactors Addendum No. 1 Multiple Enrichment Cores With Axial Gadolinium." NEDO-10527, Supplement 1. General Electric Company, July 1972.

Section 9

QUALITY ASSURANCE

Although the HWC system is non-nuclear safety related, the design, procurement, fabrication and construction activities shall conform to the quality assurance provisions of the codes and standards specified herein. In addition, or where not covered by the referenced codes and standards, the following quality assurance features shall be established.

9.1 SYSTEM DESIGNER AND PROCURER

1. Design and Procurement Document Control--Design and procurement documents shall be independently verified for conformance to the requirements of this document by individual(s) within the design organization who are not the originators of the design and procurement documents. Changes to design and procurement documents shall be verified or controlled to maintain conformance to this document.

2. Control of Purchased Material, Equipment and Services--Measures shall be established to ensure that suppliers of material, equipment and construction services are capable of supplying these items to the quality specified in the procurement documents. This may be done by an evaluation or a survey of the suppliers' products and facilities.

3. Handling, Storage, and Shipping--Instructions shall be provided in procurement documents to control the handling, storage, shipping and preservation of material and equipment to prevent damage, deterioration, and reduction of cleanliness.

9.2 CONTROL OF HYDROGEN STORAGE AND/OR GENERATION EQUIPMENT SUPPLIERS

In addition to the requirements in section 9.1, the system designer should audit the design and manufacturing documents of the equipment supplier to assure conformance to the procurement documents. The system designer shall specify specific factory tests to be performed which will assure operability of the supplier's equipment. The system designer or his representative should be present for the factory tests.

9.3 SYSTEM CONSTRUCTOR

1. Inspection--In addition to code requirements, a program for inspection of activities affecting quality shall be established and executed by, or for, the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. This shall include the visual inspection of components prior to installation for conformance with procurement documents and visual inspection of items and systems following installation, cleaning, and passivation (where applied).

2. Inspection, Test and Operating Status--Measures shall be established to provide for the identification of items which have satisfactorily passed required inspections and tests.

3. Identification and Corrective Action for Items for Nonconformance--Measures shall be established to identify items of nonconformance with regard to the requirements of the procurement documents or applicable codes and standards and to identify the remedial action taken to correct such items.

Appendix A

CODES, STANDARDS, REGULATIONS, AND PUBLISHED GOOD ENGINEERING PRACTICES APPLICABLE TO PERMANENT HYDROGEN WATER CHEMISTRY INSTALLATIONS

This appendix lists codes, standards, and regulations which may be applicable to specific permanent hydrogen water chemistry installations.

10 CFR 20	Standards for Protection Against Radiation
10 CFR 50.48	Fire Protection
10 CFR 50.49	Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants
10 CFR 50	Appendix A, General Design Criteria for Nuclear Power Plants, General Design Criteria 54, 55, 56, or 57
10 CFR 73.55	Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage
10 CFR 100	Reactor Site Criteria
29 CFR 1910	Labor - OSHA Health Standards
29 CFR 1910.103	Hydrogen
29 CFR 1910.104	Oxygen
40 CFR 190	Protection of Environment - Environmental Radiation Protection Standards for Nuclear Power Operations

ASME Boiler and Pressure Vessel Code, Section VIII, Pressure Vessels

ASME Boiler and Pressure Vessel Code, Section IV, Heating Boilers

ASME Boiler and Pressure Vessel Code, Section IX, Welding and Brazing Qualifications

ANSI A13.1	Scheme for the Identification of Piping Systems
ANSI B31.1	American National Standards Institute, Power Piping
ANSI B31.3	American National Standards Institute, Chemical Plant and Petroleum Refinery Piping
ANSI Z35.1	Accident Prevention Signs, Specification for
API Standard 620	Design and Construction of Large, Welded, Low-Pressure Storage Tanks, America Petroleum Institute Recommended Rules for

ASTM G63-83a	Evaluating Nonmetallic Materials for Oxygen Service
ASTM G88-84	Designing Systems for Oxygen Service
AWS D1.1	Structural Welding Code
NFPA 50	Bulk Oxygen Systems
NFPA 50A	Gaseous Hydrogen Systems at Consumer Sites
NFPA 50B	Liquified Hydrogen Systems at Consumer Sites
NFPA 53M	Fire Hazards in Oxygen-Enriched Atmospheres
NFPA 70	National Electrical Code
NFPA 78	Lightning Protection Code

Compressed Gas Association G-4, Oxygen

Compressed Gas Association G-4.1, Cleaning Equipment for Oxygen Service

Compressed Gas Association G-4.3, Commodity Specification for Oxygen

Compressed Gas Association G-4.4, Industrial Practices for Gaseous Oxygen Transmission and Distribution Piping Systems

Compressed Gas Association G-5, Hydrogen

Compressed Gas Association G-5.3, Commodity Specification for Hydrogen

Compressed Gas Association P-12, Safe Handling of Cryogenic Liquids

U.S. Army Technical Manual TM5-1300

U.S. Department of Transportation Specification 3A, 3AA, 3AX, 3AAX

U.S. Nuclear Regulatory Commission Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants"

U.S. Nuclear Regulatory Commission Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transporation Routes Near Nuclear Power Plants"

U.S. Nuclear Regulatory Commission Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable (ALARA)"

U.S. Nuclear Regulatory Commission Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable"

U.S. Nuclear Regulatory Commission Branch Technical Position CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants"

U.S. Nuclear Regulatory Commission NUREG-0800, "Standard Review Plan For The Review of Safety Analysis Reports For Nuclear Power Plants."

Appendix B

SEPARATION DISTANCES RECOMMENDED FOR HYDROGEN STORAGE TO PREVENT DAMAGE TO NUCLEAR POWER PLANT STRUCTURES FROM HYDROGEN EXPLOSION

by

R. P. Kennedy

RPK

Structural Mechanics Consulting

Robert P. Kennedy

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August 1, 1986

Mr. Robin Jones
Electric Power Research Institute
3412 Hillview Avenue
P.O. Box 10412
Palo Alto, CA 94303

Re: Report on Hydrogen Storage Separation Distances

Dear Mr. Jones:

Enclosed is my final report on the subject material entitled "Separation Distances Recommended for Hydrogen Storage to Prevent Damage to Nuclear Power Plant Structures from Hydrogen Explosion". This report provides screening material for conservatively estimating separation distances for hydrogen storage. The approach is conservative because:

a) Concrete walls are assumed to be 18-inches thick and lesser distances coud be used for thicker walls.

b) A minimum amount of tensile reinforcement is assured while greater reinforcement would reduce separation distance.

In cases where the actual separation distances are less than those recommended, a dynamic blast capacity analysis may justify lesser separation distances.

I wish to thank you for this opportunity to have been of service to EPRI. This submittal completes my workscope under EPRI Agreement CSA86-28.

Very truly yours,

Robert P. Kennedy

Robert P. Kennedy

cc. Mr. Robert Bazarian

SEPARATION DISTANCES RECOMMENDED FOR HYDROGEN STORAGE TO PREVENT DAMAGE TO NUCLEAR POWER PLANT STRUCTURES FROM HYDROGEN EXPLOSION

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June 1986

Prepared for

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SEPARATION DISTANCES RECOMMENDED FOR HYDROGEN STORAGE TO PREVENT DAMAGE TO NUCLEAR POWER PLANT STRUCTURES FROM HYDROGEN EXPLOSION

R. P. Kennedy

June 1986

1. Introduction

One method to arrest pipe cracking and pipe crack growth due to intergranular stress corrosion of austenitic stainless steel piping consists of maintaining good water chemistry and adding hydrogen to the feedwater. Addition of hydrogen decreases the oxidizing power of the water and reduces its aggressiveness toward austenitic stainless steel piping materials. However, this approach requires significant hydrogen storage at the plant site in gaseous or liquid hydrogen storage systems. Although highly unlikely, with such storage one can postulate a storage vessel or storage tank rupture possibly leading to hydrogen explosion with blast waves propagating outward from the vessel or tank rupture site. The blast wave characteristics from such a postulated explosion can be expressed in terms of an equivalent TNT explosion. The Electric Power Research Institute has conservatively recommended (Ref. 1) that the following TNT equivalences be used:

Gaseous Hydrogen Storage

1000 Standard Cubic Feet Hydrogen (SCF) = 27.1 lbs. TNT (1)

Liquid Hydrogen Storage

1 Gallon Liquid Hydrogen = 1.37 lbs. TNT (2)

Using these equivalencies, the question addressed in this report is, how far do these gaseous hydrogen storage vessels or liquid hydrogen storage tanks have to be separated from important structures at a nuclear power plant to prevent structure walls from being seriously damaged if a hydrogen explosion does occur.

The purpose of this report is to provide conservative, bounding separation distances which require a minimum of knowledge about the material properties and static or dynamic capacities of the structure walls. If these separation distances are maintained, then no blast resistance evaluation of the walls is required. If storage vessels or tanks are to be located closer than these separation distances to important structural walls, then a blast resistance evaluation is required for such walls. Because of conservatism on the bounding criteria, such blast resistance evaluations are likely to show many actual walls to be safe at lesser standc' distances than those recommended herein. Recommendations in report are for reinforced concrete or masonry walls which are at least 8 inches thick. Structures with light-gage metal paneling walls and exposed metal tanks are not covered in this report. Such structures will have to be individually assessed.

First, this report provide ration distances based upon the recommendation of the British Explosives Storage and Transport Committee (E.S.T.C.) for the protection of public safety against blast (Ref. 2). These recommendations are based upon well-documented explosion experiments together with damage records from accidental explosions and enemy bombing in the United Kingdom. Their recommendations are intended to prevent more than a 5% probability of ious structural damage to standard British unreinforced brick dwelling houses. Certainly no 8-inch thick reinforced concrete or masonry wall at a nuclear power plant would have lesser blast capacity than these unreinforced brick dwelling houses. Therefore, it is my opinion that the British E.S.T.C. recommended separation distances can be safely used for all 8-inch or greater thickness reinforced concrete or reinforced masonry walls at nuclear plants without any investigation being required.

If one plans to use separation distances less than those recommended by the British E.S.T.C., then some additional knowledge is necessary. External walls of safety-class structures for most nuclear power plants have been designed to be resistant to the effects of tornadoes. The required level of resistance depends upon the tornado zone in which the plant is located. For Tornado Zones I thru III, the minimum required static lateral load capacities, P_s, are (Ref. 4):

Tornado Zones	Minimum Static Capacities P_s (psi)
Zone I	3.0
Zone II	2.25
Zone III	1.5

In addition, these walls are also designed to withstand impacts from very conservatively hypothesized tornado generated missiles and have certain minimum requirements for reinforcing steel. Any wall meeting such design requirements will have significantly greater blast resistance capacity than that envisioned by the British E.S.T.C. recommended separation distances. NUREG/CR-2462 (Ref. 3) makes conservative, bounding separation distance recommendations for such walls. The recommended separation distance is a function of the static pressure capacity, P_s, of the wall and the permissible level of inplastic deformation in terms of ductility ratio, μ. The recommendations assume that the wall is reinforced concrete with at least 0.2% tensile steel reinforcement (which represents a lower bound on the amount of tensile steel than I have ever seen reported for an external wall of a nuclear power plant safety-class structure) and is at least 18 inches thick (the minimum thickness required to withstand the hypothesized tornado-generated missiles). So long as these assumptions are met, the NUREG/CR-2462 recommended separation distances may be used.

The NUREG/CR-2462 (Ref. 3) formulas for recommended separation distances were developed for external blast explosions with a TNT-equivalence of greater than 40,000 lbs. These formulas become excessively conservative for explosions of less than 40,000 lbs. TNT-equivalence. The TNT-equivalence of any explosion associated with gaseous hydrogen storage is certainly expected to be less than 400 lbs., which corresponds to almost 15,000 SCF of hydrogen gas storage, so that the Reference 3 formulas would be excessively conservative for these cases. The TNT-equivalence for liquid storage capacities ranging from 1,000 gallons to 100,000 gallons of liquid hydrogen is 1,370 lbs. to 137,000 lbs. TNT-equivalence. For the smaller such tanks (less than 25,000 gallons), use of the Reference 3 formulas for separation distances would be excessively conservative.

Therefore, this report modifies the separation distance (standoff distance) formulas of Reference 3 to extend them down to blasts with much lesser TNT-equivalence. For large TNT-equivalence explosions such as those considered in Reference 3, wall damage correlates mostly with the peak incident overpressure while for small TNT-equivalence explosion, wall damage correlates mostly with the total incident positive impulse of the blast wave. The separation distance formulas provided in this report cover the complete spectrum and are based upon computed reinforced concrete wall responses for explosions with TNT-equivalences ranging from 40 lbs. to 1×10^7 lbs. The methodology used and all assumptions are identical to those described in Reference 3. The conservative bounding separation distances are applicable for reinforced concrete walls with the following properties:

Wall Thickness:	$t \geq 18$ inches
Tensile Steel Factor:	$\left(\frac{\rho\ fy}{100\%}\right) \geq 0.12$ ksi & 0.3 ksi
Static Pressure Capacity:	$1.5\text{psi} \leq P_s \leq 4.5\text{psi}$
Permissible Ductility:	$\mu = 1.0,\ 3.0,\ 5.0$

For large TNT-equivalence explosions (greater than 40,000 lbs.), the conservatively required separation distances are only mildly sensitive to wall thickness and one may use the recommended separation distance formulas for wall thicknesses down to 15 inches with only slight unconservatism. However, for small TNT-equivalence explosions (less than 4,000 lbs.), conservatively recommended separation distances are sensitive to wall thickness. All results are for 18-inch thick walls and are conservative for thicker walls. Recommended separation distances for all TNT-equivalence explosions are very sensitive to the tensile steel factor, $\left(\frac{\rho\ fy}{100\%}\right)$, where ρ is the percentage of tensile steel reinforcement and fy is the steel yield strength. Recommended separation distances are presented for the cases of $\left(\frac{\rho\ fy}{100\%}\right) \geq$ 0.12ksi and 0.3 ksi which correspond to tensile steel yield strengths of 60ksi and tensile steel percentages of 0.2% and 0.5%, respectively. The $\left(\frac{\rho\ fy}{100\%}\right) \geq$ 0.12ksi case may always be used since all external concrete walls of nuclear plants will have at least this tensile steel factor. The $\left(\frac{\rho\ fy}{100\%}\right) \geq$ 0.3ksi case can only be used when one has verified that the tensile steel factor does exceed 0.3ksi. For large TNT-equivalence explosions, the recommended separation distances are sensitive to the static pressure capacity, P_s, of the wall. This term is included in the separation distance formula. It is believed that the recommended formula should be valid over a wide range of static pressure capacities. However, this formula has only been checked against computed wall responses for walls with static pressure capacities between 1.5 and 4.5psi. One should exercise some caution when extrapolating the recommended formula beyond this range. It is always conservative to underestimate the static pressure capacity. The conservatively recommended separation distances are not sensitive to the concrete strength. All computations were based on a minimum concrete compressive strength of 4,000psi. Lastly, the recommended separation distances are sensitive to the permissible level of inelastic deformation of the wall as defined by the permissible ductility factor, μ. A ductility factor of 1.0 represents elastic response while ductility factors of 3.0 and 5.0 represent total wall deflections of 3 and 5 times the wall yield deflection, respectively. For external blast loads from very unlikely explosions, I recommend the use of the $\mu = 3$ based separation distances. The $\mu = 1$ and $\mu = 5$ separation distances are presented for conditions where one wishes to exercise greater or lesser conservatism.

The separation distance formulas in this report may be used in the following manner:

1. First check against the British E.S.T.C. recommended separation distance (Equation 3). If actual distances exceed this recommended distance, then no further checks are necessary. Otherwise, proceed to Step 2.

2. Determine if wall is at least 18 inches thick and estimate minimum static capacity based upon the tornado zone for which the wall has been designed (for Tornado Zone I, the minimum P_s = 3.0psi). For walls which are at least 18 inches thick, determine conservative separation distance from Equation 7 for $\left(\frac{\rho \ fy}{100\%}\right) \geq 0.12$ksi, $\mu = 3.0$, and appropriate minimum P_s case. If acceptable, then no further work is needed. Otherwise, proceed to Step 3.

3. Determine whether $\left(\frac{\rho \ fy}{100\%}\right) \geq 0.30$ksi. If so, repeat Step 2 with this higher tensile steel factor so as to obtain a lesser separation distance. If acceptable, no further work is needed. Otherwise, proceed to Step 4.

4. Perform static pressure capacity, P_s, calculation for wall and use this actual static capacity in Step 2 or 3 in lieu of the minimum capacity estimate which was based on the design tornado zone. However, one should be cautious about extrapolating Equation 7 for static capacities substantially greater than 4.5psi since this formula has not been checked for such cases. If the resultant separation distance is acceptable, no further work is needed. Otherwise, proceed to Step 5.

5. Perform dynamic blast capacity analysis in accordance with the procedures of Reference 3 in order to remove all unnecessary conservatism from the preceding bounding separation distance approaches.

Each step in the above procedure requires additional work but removes some conservatism associated with the previous step.

This report is intended to serve as a supplement to the previously published EPRI "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations" (Ref. 1). That previous EPRI report presents some recommendations for separation distances for gaseous hydrogen storage systems and liquid hydrogen storage tanks through Figures 4.2, 4.3, and 4.6 of that report (Ref. 1). Representative recommendations from this report have been directly overlaid onto Figures 4.2, 4.3, and 4.6 of Reference 1. For this purpose, the original EPRI figures have been reproduced in this report as Figures 2 thru 4.

2. British E.S.T.C. Recommended Separation Distances

The British E.S.T.C. has recommended (Ref. 2) the following equation for determining public safety separation distances in order to achieve no more than a 5% probability of serious structural damage to typical unreinforced brick British dwelling house construction:

$$R = \frac{56\ W^{1/3}}{\left[1 + \left(\frac{7,000}{W}\right)^2\right]^{1/6}} \tag{3}$$

where R is the separation distance in feet and W is the TNT-equivalent explosive yield in lbs. It should be noted that Equation 3 does not represent an abrupt demarcation between "safe" and "unsafe" performance of such structures. As separation distances are substantially reduced below those given by Equation 3, the level of damage gradually increases. In fact the constant, K = 56, in Equation 3 must be reduced to K = 14 to correspond to situations where on the average 50 to 75% of the external brickwork is destroyed or rendered unsafe and requiring demolition. Use of a K = 56 results in separation distances four times as great as that obtained for K = 14 for which very severe damage is highly likely. Thus, Equation 3 has built-in conservatism. Even so, empirical evidence indicates that about 5% of the time serious structural damage requiring wall replacement does occur to these typical unreinforced brick structures at distances exceeding those given by Equation 3. Even with a K = 140 or 2.5 times greater distance than given by Equation 3, one obtains 10% window glass breakage. Thus, the empirical evidence shows a wide scatter in separation distances to prevent differing levels of damage.

In my opinion, Equation 3 represents a generally safe separation distance for heavy load bearing walls which have very low static lateral pressure capacity as do unreinforced load bearing brick walls. Such walls tend to be kept in place under blast loads by their large inertial weight forces and not by their static lateral pressure capacity which often is low. Any heavy load bearing wall which has been designed for significant lateral forces such as those due to earthquakes or those due to tornado side-on pressure loads should have greater blast capacity than these unreinforced load bearing brick walls. However, for large structures, such as those at nuclear power plants, the blast loads imposed on the walls have as much as twice the total impulse per unit area as that on small structures, such as those British dwelling houses, because it takes longer for reflected pressures to clear around large structures. Even recognizing this fact, I feel certain that Equation 3 represents a very conservative separation distance criterion for any heavy reinforced concrete or reinforced masonry wall that is at least 8 inches thick since these walls do have static lateral pressure capacity due to the reinforcing steel. In my opinion, it is inconceivable that a thick, reinforced, load-bearing wall could be seriously damaged at standoff distances given by Equation 3.

The free-field blast wave parameters that primarily influence structural response are peak incident overpressure, P_{so}, and the total positive incident impulse, I_{so}. For a TNT-equivalent surface burst explosion, Figure 1 (taken from Ref. 5) presents the relationship between P_{so}, I_s, and the scaled range, Z, where:

$$Z = \frac{R}{W^{1/3}} \tag{4}$$

$$I_{so} = I_s W^{1/3} \tag{5}$$

Using Figure 1 together with Egns (4) and (5), one can convert the separation distance equation (Egn 3) into a plot of peak incident overpressure, P_{so}, versus total positive incident impulse, I_{so}, that corresponds to less than a 5% probability of serious structural damage to these unreinforced brick load bearing walls. The process is as follows:

1. Assume an explosion yield, W, in lbs. and use Egn (3) to compute the separation distance, R, and use Egn (4) to obtain the scaled range, Z.

2. Enter Figure 1, with the scaled range, Z, and read off the peak incident overpressure, P_{so}, and scaled total positive incident impulse, I_s. Then use Egn (5) to obtain the total positive incident impulse, I_{so}.

3. Repeat Steps 1 and 2 for the full range of yields, W, so that a complete table of the interaction between P_{so} and I_{so} that leads to 5% probability of serious structural damage to unreinforced brick load bearing walls can be prepared based upon Egn (3).

Such an interaction between P_{so} and I_{so} leading to 5% probability of serious structural damage to unreinforced, load bearing brick walls is provided in Table 1.

Table 1 indicates that for explosive yields less than 4,000 lbs. TNT, this level of damage corresponds to a total positive incident impulse of 31.5psi-ms and is completely independent of the peak incident overpressure. Basically, it takes this much impulse to overcome the stability provided by the inertial weight of the wall and until at least this total positive impulse is provided the wall will not be damaged. However, for very large explosive yields greater than 16,000 lbs. TNT, this level of damage is found to correlate with a peak incident overpressure of 0.75psi irrespective of the total positive impulse. Only between 4,000 lbs. and 16,000 lbs. is there an interaction effect between P_{so} and I_{so} corresponding to the observed level of damage, and even here the interaction is not large.

As a good approximation, one can state that the British experience indicates about a 5% probability of serious structural damage to unreinforced brick load bearing walls whenever <u>both</u> of the following are exceeded:

$$\left.\begin{array}{l} P_{so} \geq 0.75\text{psi} \\ \quad\text{and} \\ I_{so} \geq 31.5\text{psi-ms} \end{array}\right\} \qquad (6)$$

For other structural materials (such as reinforced concrete walls) or other levels of damage, similar damage relationships can be produced which show that damage depends upon exceeding both an incident overpressure and a total impulse.

Figure 2 plots the relationship between incident impulse and peak incident overpressure corresponding to 5% probability of serious structural damage of load bearing brick walls. Note that for overpressure greater than 1psi, this plot is simply a straight line at 31.5psi-ms. Also shown in Figure 2 is the EPRI damage threshold line from Reference 1. Such a line is significantly higher than that corresponding to the British E.S.T.C. recommended separation distances. Lastly, Figure 2 shows several lines corresponding to the recommended separation distances for 18-inch or greater thickness reinforced concrete walls permitted to deform to a ductility factor of 3.0.

3. Recommended Separation Distances for 18-inch and Greater Thickness Reinforced Concrete Walls

As mentioned in Section 1, this report extends the separation distance formulas presented in NUREG/CR-2462 (Ref. 3) down to cases with lesser explosive yields, W. Reference 3 studied both 18-inch and 24-inch thick walls with various support conditions, and various tensile steel factors, $\left(\frac{\rho\ f_y}{100\%}\right)$. The 18-inch thick walls with simple supported spans were found to control the recommendations of Ref. 3 for a conservative bounding separation distance. Properties of these walls were tabulated in Table A1 of Reference 3. However, the blast capacity analyses of Reference 3 which led to its separation distance recommendations only went down to 40,000 lbs. of TNT-equivalent yield. Therefore the recommendations of Reference 3 are only appropriate when the peak incident overpressure, P_{so}, controls the wall behavior and are too conservative for lesser TNT-equivalent yields. To extend the Reference 3 separation distance equation down to lower yields, the following 18-inch wall thickness, simple-supported span cases of Table A-1 of Reference 3 have been analyzed at lesser yields:

Tensile Steel Factor, $\left(\frac{p\ f_y}{100\%}\right)$:	0.12ksi, 0.30ksi
Static Pressure Capacity, P_s:	1.5psi, 3.0psi, 4.5psi
Permissible Ductility, μ:	1.0, 3.0, 5.0
Explosive Yield, W:	40 lbs., 100 lbs., 400 lbs., 1,000 lbs., 4,000 lbs., 10,000 lbs.

For each of these wall cases, results already exist in Reference 3 for higher explosive yields of approximately 40,000 lbs., 400,000 lbs., and 1×10^7 lbs. These results were also used to come up with the separation distance recommendations of this report.

The methodology used and all assumptions are fully described in Reference 3 and will not be repeated herein. At these lesser yields, the reflected pressure, P_r, and reflected impulse, I_r, exceed 2 times the peak incident overpressure, P_{so}, and total positive incident impulse, I_{so}. Therefore, for these lower yields, P_r and I_r were obtained directly from Figure 1 rather than assume they are 2 times the incident values as was done in Reference 3.

It was found that a separation distance formula with the format of Egn 3 could conservatively approximate (never more than 5% unconservative) computed separation distances for all of the cases studied. The resultant separation distance formula is:

$$R = \frac{K\ W^{1/3}}{\left[1 + \left(\frac{(K/B_\mu)^4}{W}\right)^2\right]^{1/8}} \tag{7}$$

where:

$$K = \frac{\phi_\mu}{(P_s)^{2/3}} \tag{8}$$

and:

μ	$\left(\frac{\rho\ f_y}{100\%}\right) = 0.12ksi$		$\left(\frac{\rho\ f_y}{100\%}\right) = 0.30ksi$	
	f_μ	B_μ	f_μ	B_μ
1.0	87	5.2	80	2.7
3.0	54	2.8	50	1.45
5.0	51	2.2	44	1.2

The use of Eqn (7) requires one to estimate:

1) Minimum static capacity, P_s

2) Tensile steel factor, $\left(\frac{\rho\ f_y}{100\%}\right)$

3) Permissible wall ductility, μ

Some representative cases are:

P_s = 1.5psi ; $\left(\frac{\rho\ f_y}{100\%}\right)$ = 0.12ksi ; μ = 3.0

$$R = \frac{41\ W^{1/3}}{\left[1 + \left(\frac{47,000}{W}\right)^2\right]^{1/8}}$$

P_s = 4.5psi ; $\left(\frac{\rho\ f_y}{100\%}\right)$ = 0.12ksi ; μ = 3.0

$$R = \frac{20\ W^{1/3}}{\left[1 + \left(\frac{2,500}{W}\right)^2\right]^{1/8}}$$

P_s = 3.0psi ; $\left(\frac{\rho\ f_y}{100\%}\right)$ = 0.12ksi ; μ = 1.0

$$R = \frac{42\ W^{1/3}}{\left[1 + \left(\frac{4,200}{W}\right)^2\right]^{1/8}}$$

P_s = 3.0psi ; $\frac{\rho\ f_y}{100\%}$ = 0.30ksi ; μ = 3.0

$$R = \frac{24\ W^{1/3}}{\left[1 + \left(\frac{75,000}{W}\right)^2\right]^{1/8}}$$

Tables 2 thru 5 present minimum required separation distances based upon Eqn (7) for various explosive yields, W, ranging from 40 lbs. to 1×10^7 lbs. for these 4 representative cases, respectively. Also shown in these tables are the bounding separation distances together with the peak incident overpressure and total positive impulse capacities computed by blast response calculations for these walls using the bounding capacity approach of Reference 3. One should note the very close agreement between the separation distances provided by Eqn (7) with the bounding separation distances obtained from bounding blast capacity computations using the methodology of Reference 3. In general, Equation 7 introduces slight conservatism and is never more than 5% unconservative versus results from these conservative bounding blast capacity computations. Again, it should be noted that these computations are for 18-inch thick walls and can be conservatively used for walls of greater thickness.

Tables 2 through 5 also provide interaction data between the peak incident overpressure, P_{SO}, and the total positive incident impulse, I_{SO}, corresponding to each of the four cases considered. It has previously been recommended that for design evaluations one should use a ductility factor, μ, of 3.0 for external blast loads. Tables 2, 3, and 5 provide interaction data for $\mu = 3.0$. Interaction data for these three cases is plotted on Figure 2 so that these cases can be compared with both the British E.S.T.C. recommendations for unreinforced brick load bearing walls and with the EPRI (Ref. 1) damage curve recommendations. Note that at a ductility of 3.0, the 18-inch thick wall with a tensile steel factor, $\left(\frac{\rho\ f_y}{100\%}\right)$, of 0.12ksi has a somewhat lower minimum impulse capacity than the EPRI damage curve. Furthermore, even with a static capacity of 4.5psi this wall has less minimum peak incident overpressure capacity than does the EPRI damage curve. This wall would have to have a static pressure capacity of about 10psi in order to have a minimum peak incident overpressure capacity as high as that shown by the EPRI damage curve. The 18-inch thick wall with a tensile steel factor, $\left(\frac{\rho\ f_y}{100\%}\right)$, of 0.30ksi has a minimum total positive incident impulse capacity higher than the EPRI damage curve. This curve shows the importance of increased tensile steel levels on the minimum impulse capacity.

4. Minimum Required Separation Distances for Gaseous Hydrogen Storage Systems

Using the conversion from gaseous hydrogen to an equivalent weight of TNT explosion given by Egn 1, the British E.S.T.C. recommended separation distances for unreinforced brick walls (Egn 3), or the recommended separation distances for 18-inch thick reinforced concrete walls (Egn 7) can be estimated as a function of storage vessel size. Figure 3 presents recommended separation distances versus vessel size for unreinforced brick walls (British E.S.T.C.) and for 18-inch thick reinforced concrete walls with $P_s \geq 1.5$psi and a permissible ductility factor, of 3.0. Also shown in Figure 3 is the EPRI recommended separation distance (Ref. 1).

5. Minimum Required Separation Distances for Liquid Hydrogen Storage Tanks

Similarly, separation distances versus storage tank size are shown in Figure 4 for liquid hydrogen storage. These distances are based on the conversion from liquid hydrogen to an equivalent TNT explosive weight given by Egn (2). Essentially all of the BWR plants considering liquid hydrogen storage vessels are in Tornado Zone I for which the minimum static wall capacity, P_s, must be as least 3.0psi. Therefore, the curves in Figure 4 for at least 18-inch thick reinforced concrete walls are based on P_s = 3.0psi and a permissible ductility factor, μ, of 3.0. The 18-inch thick reinforced concrete wall separation distance curve for a minimum tensile steel factor of 0.12ksi requires greater separation distances than the EPRI recommended separation distances (Ref. 1) for all storage tank sizes greater than 1,000 gallon. However, when $\left(\frac{\rho \; f_y}{100\%}\right)$ = 0.30ksi the EPRI recommended separation distances are conservative for storage tank sizes less than 10,000 gallon.

6. Summary

This report presents equations for estimating the recommended separation distance as a function of TNT-equivalent yield, W, for unreinforced load bearing brick walls (Egn 3) and for 18-inch thick reinforced concrete walls (Egn 7). A procedure is recommended (Section 1) for the use of these formulas to assess separation distances from safety class structures for hydrogen storage. Figures 3 and 4 provide plots of recommended separation distances versus storage vessel or tank size for gaseous hydrogen and liquid hydrogen storage, respectively. Recommended separation distances presented in this report are compared with those previously recommended by EPRI (Ref. 1).

Table 1.

Interaction Between P_{so} and I_{so} Corresponding to 5% Probability of Serious Structural Damage for Unreinforced Brick Load Bearing Walls in Typical British Dwelling Construction.

Yield W, lbs.	R, ft	P_{so}, psi	I_{so}, psi-ms
10	13.5	25.5	31.5
40	34.3	9.5	31.5
100	63.1	5.4	31.5
400	159	2.7	31.5
1,000	292	1.8	31.5
4,000	703	1.0	31.5
8,000	1,020	0.82	34.4
16,000	1,370	0.76	37.8
40,000	1,910	0.75	51.3
100,000	2,600	0.75	69.6
400,000	4,130	0.75	111.0

Table 2.

Separation Distance and Blast Load Capacity for 18-inch Thick Reinforced Concrete Walls with P_s = 1.5psi, $\left(\frac{\rho\ f_y}{100\%}\right)$ = 0.12ksi and μ = 3.0

Yield W, lbs.	Blast Capacity Computations			(Egn 7) R, ft
	R, ft	P_{so}, psi	I_{so}, psi-ms	
40	23	22	48	24
100	40	14	52	41
400	92	6.4	56	92
1,000	155	4.5	60	155
4,000	360	2.5	65	350
10,000	630	1.8	70	600
43,000	1,190	1.5	105	1,310
396,000	2,940	1.2	202	3,020
1×10^7	8,880	1.15	549	8,880

Table 3.

Separation Distance and Blast Load Capacity for 18-inch Thick Reinforced Concrete Walls with P_s = 4.5psi, $\left(\frac{\rho\, f_y}{100\%}\right)$ = 0.12ksi and [illegible] = 3.0

Yield W, lbs.	Blast Capacity Computations			(Eqn 7)
	R, ft	P_{so}, psi	I_{so}, psi-ms	R, ft
40	23	22	48	24
100	39	15	54	41
400	88	6.8	60	92
1,000	150	4.7	62	155
4,000	280	3.5	80	300
10,000	405	3.3	104	420
43,000	665	3.25	168	690
396,000	1,470	3.0	338	1,460
1×10^7	4,460	2.85	969	4,270

Table 4.

Separation Distance and Blast Load Capacity for 18-inch Thick Reinforced Concrete Walls with P_s = 3.0psi, $\left(\frac{\rho\, f_y}{100\%}\right)$ = 0.12ksi and [illegible] = 1.0

Yield W, lbs.	Blast Capacity Computations			(Eqn 7)
	R, ft	P_{so}, psi	I_{so}, psi-ms	R, ft
40	44	6.3	25.5	45
100	78	4.0	25.5	76
400	175	2.3	27.5	170
1,000	300	1.65	33	290
4,000	560	1.35	43	600
10,000	800	1.30	58	880
43,000	1,330	1.25	91	1,470
396,000	2,940	1.2	202	3,080
1×10^7	8,880	1.15	549	9,050

Table 5.

Separation Distance and Blast Load Capacity for 18-inch Thick Reinforced Concrete Walls with $P_s = 3.0$psi, $\left(\frac{\rho\, f_y}{100\%}\right) = 0.30$ksi and $\mu = 3.0$

Yield W, lbs.	Blast Capacity Computations			(Eqn 7) R, ft
	R, ft	P_{so}, psi	I_{so}, psi-ms	
400	48	24	107	48
1,000	80	16	122	82
4,000	175	8.5	136	185
10,000	310	5.0	142	310
43,000	580	3.8	193	700
396,000	1,600	2.6	314	1,750
1×10^7	5,390	2.25	820	5,170

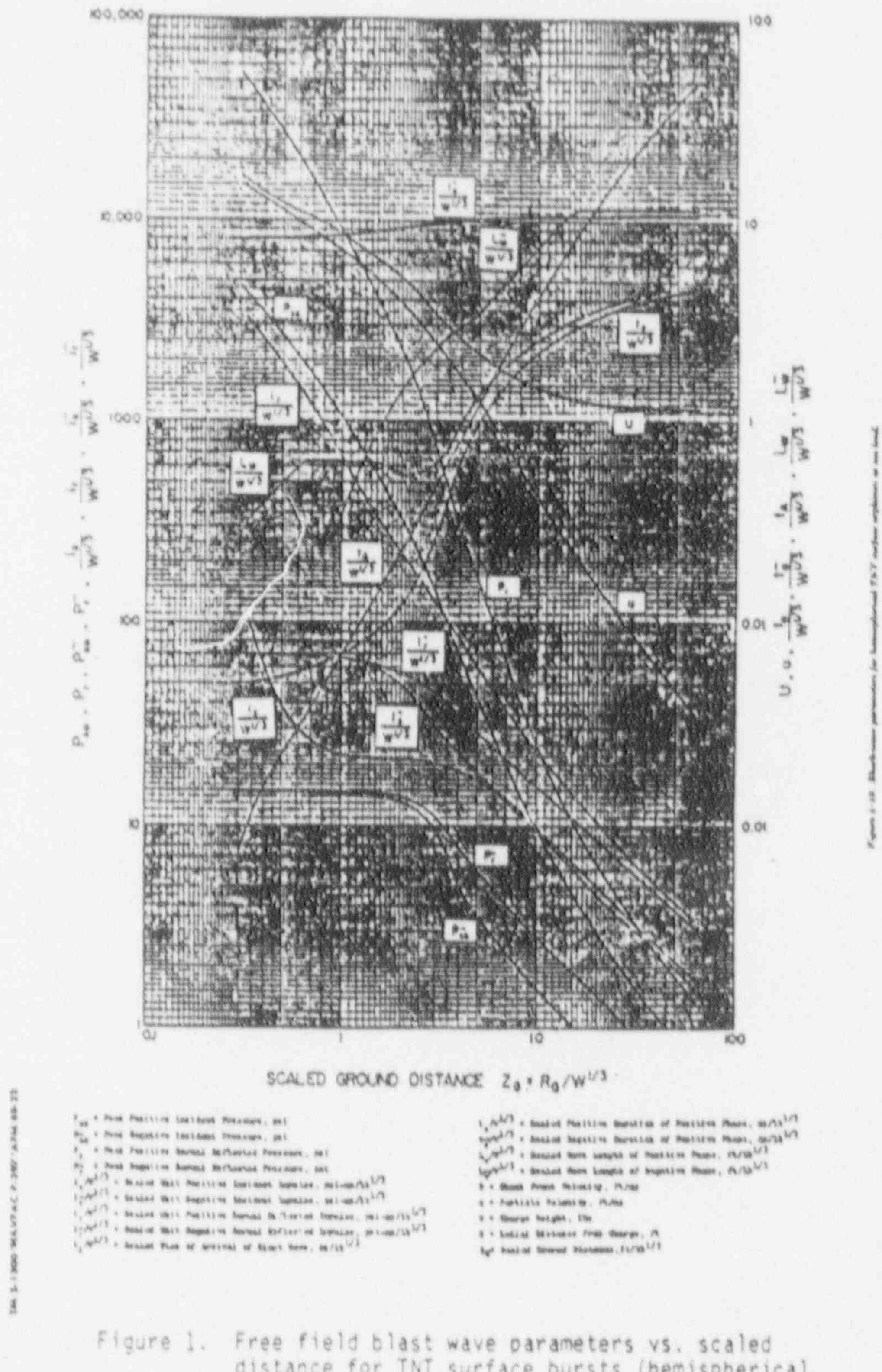

Figure 1. Free field blast wave parameters vs. scaled distance for TNT surface bursts (hemispherical charges) [5]

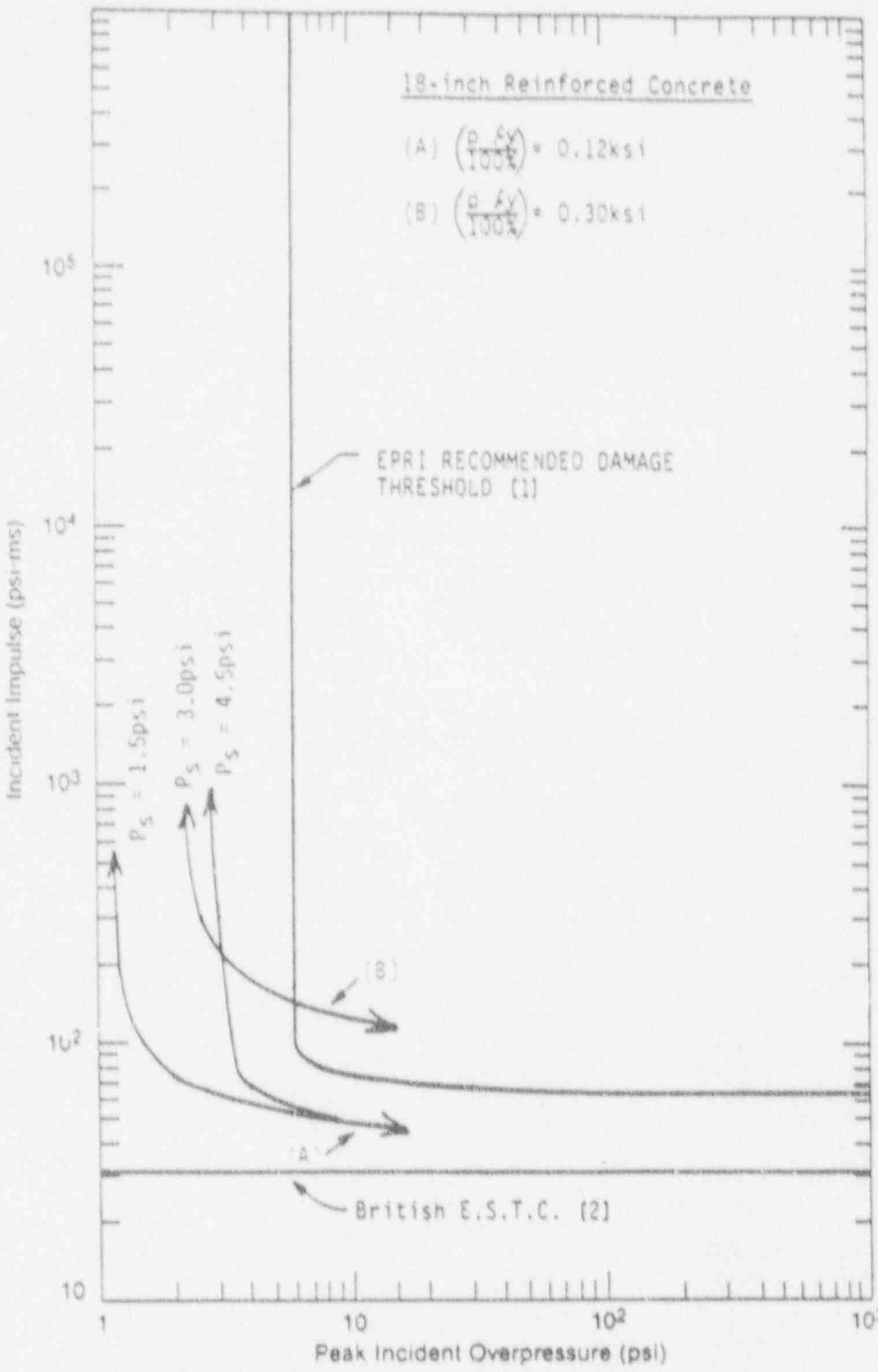

Figure 2. Relationship between incident impulse and peak incident overpressure for given damage level

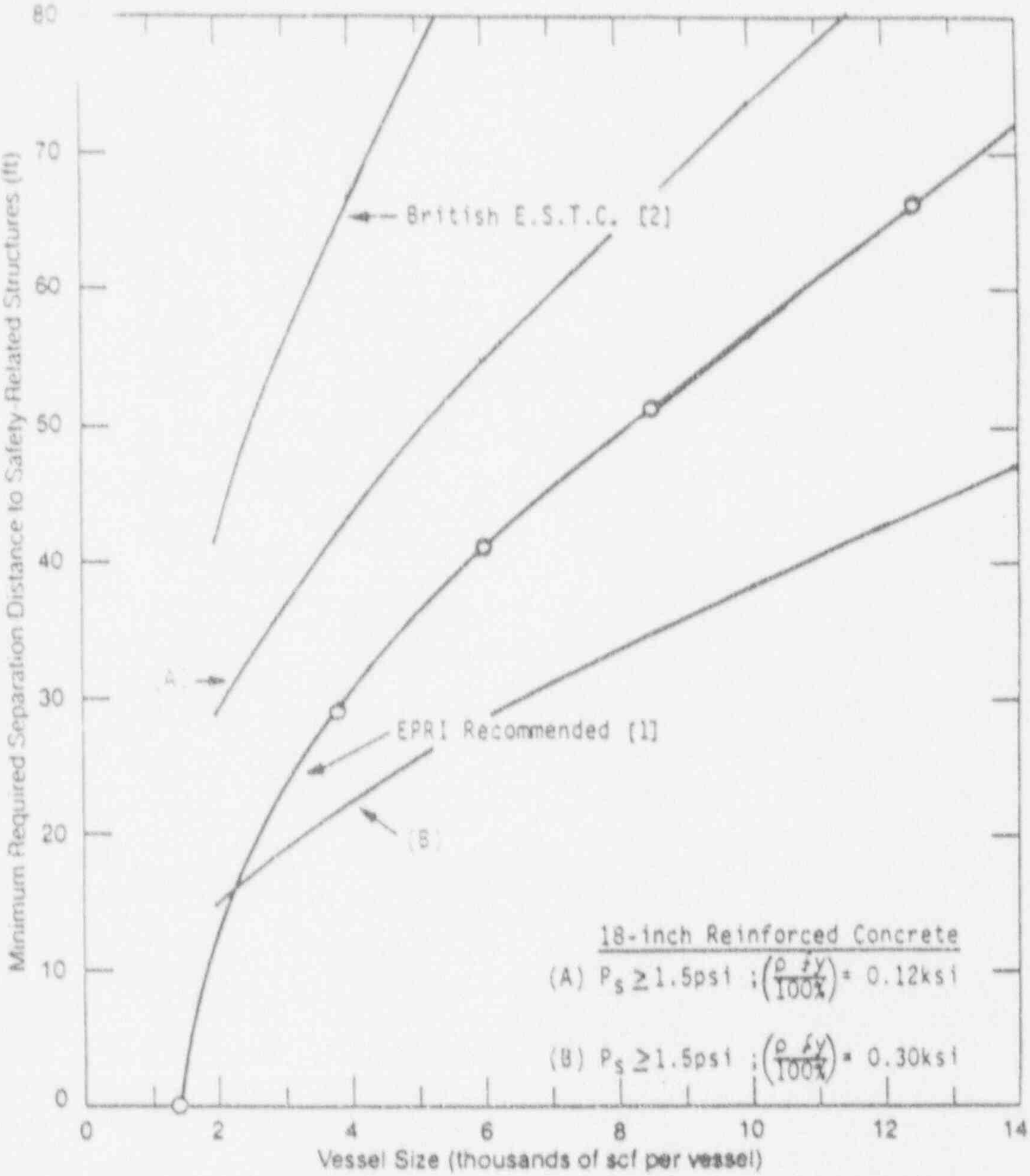

Figure 3. Minimum required separation distance to safety-related structures vs. vessel size for gaseous hydrogen storage system.

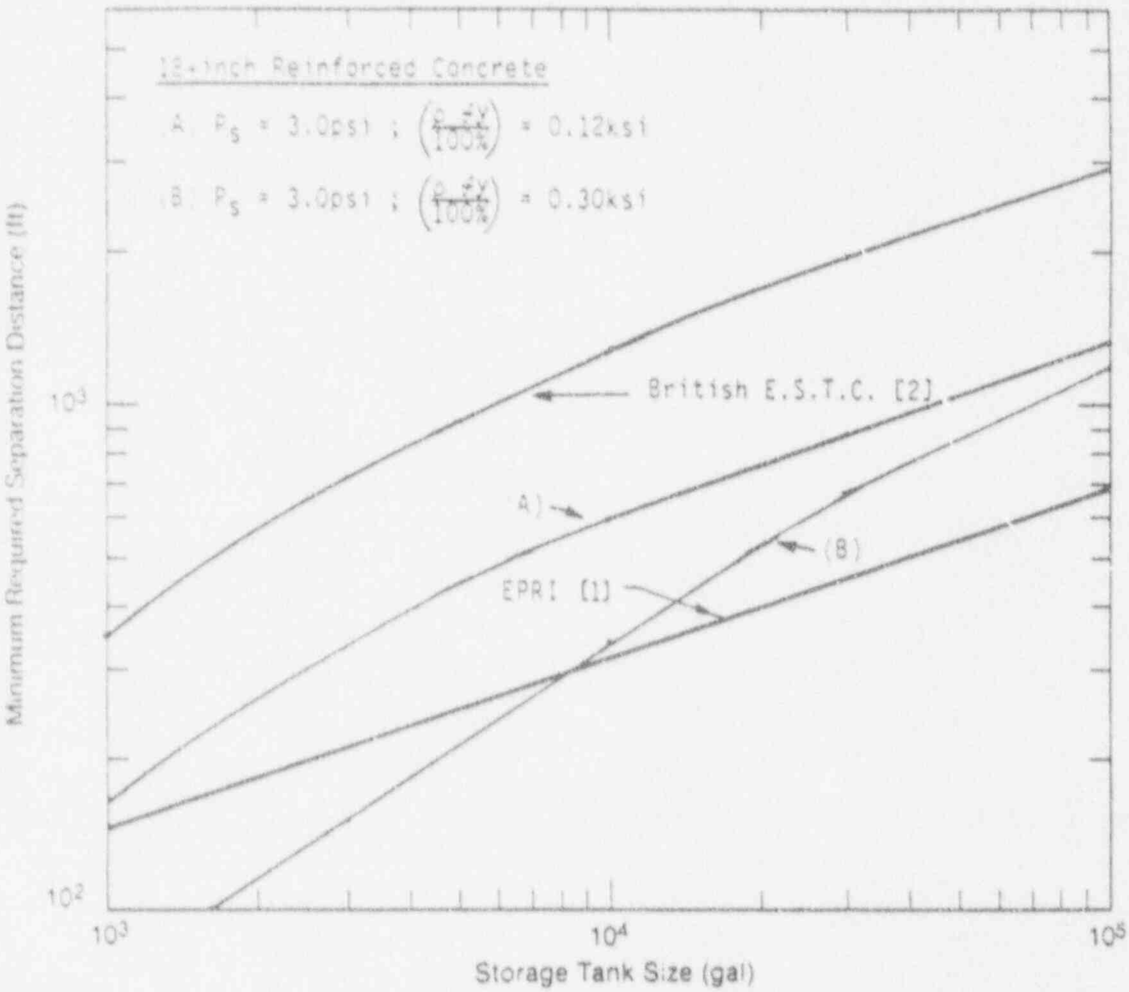

Figure 4. Minimum required separation distance vs. liquid hydrogen storage tank size for instantaneous release of entire tank contents and explosion at tank site F weather stability.

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Guidelines for Permanent BWR Hydrogen Water Chemistry Installations—1987 Revision

Prepared by
Electric Power Research Institute
Palo Alto, California

2.3 GAS INJECTION SYSTEMS

2.3.1 Hydrogen Injection System

The hydrogen injection system includes all flow control and flow measuring equipment and all necessary instrumentation and controls to ensure safe, reliable operation.

2.3.1.1 Injection Point Considerations. Hydrogen shall be injected at a location that provides adequate dissolving and mixing and avoids gas pockets at high points. Experience has shown that injection into the suction of feedwater or condensate booster pumps is feasible.

Injection into feedwater pumps will require hydrogen at high pressures (e.g., 150-600 psig). This may require either a compressed gas supply, compressors or a cryogenic hydrogen pump, depending on the supply option chosen. In the case of a liquid hydrogen storage system, this can also affect the sizing of the liquid hydrogen tank.

There may be pressure fluctuations in feedwater systems, depending on reactor power level and pump performance. The hydrogen addition system shall be designed to accommodate the full range of such fluctuations.

2.3.1.2 Codes and Standards. This system shall be designed and installed in accordance with OSHA standards in 29 CFR 1910.103.

Piping and related equipment shall be designed and fabricated to the appropriate edition of ANSI B31.1 or B31.3 for pressure-retaining components. Storage containers, if used, shall be designed, constructed, and tested in accordance with appropriate requirements of ASME B&PV Section VIII or API Standard 620. All components shall meet all the mandatory requirements and material specifications with regard to manufacture, examination, repair, testing, identification and certification.

All welding shall be performed using procedures meeting requirements in AWS D1.1, ANSI B31.1 or B31.3, or ASME B&PV, Section IX, as appropriate.

Inspection and testing shall be in accordance with requirements in ANSI B31.1, ANSI B31.3, or API 620, as appropriate.

System design shall also conform with pertinent portions of NUREG-0800, 10CFR50.48, Branch Technical position BTP CMEB 9.5-1, and appropriate standards and regulations referenced in this document. Appendix A provides a list of codes, standards, regulations, and published good engineering practices applicable to permanent hydrogen water chemistry installations. Each utility is responsible for identifying additional plant-specific codes and standards that may apply, such as State-imposed requirements, Uniform Building Code, ACI or AISC standards.

Piping and equipment shall be marked or identified in accordance with ANSI Z35.1.

2.3.1.3 <u>System Design Considerations</u>. Hydrogen piping from the supply system to the plant may be above or below ground. Piping below ground shall be designed for cathodic protection (or be coated and wrapped), the appropriate soil conditions such as frost depth or liquefaction, and expected vehicle loads. Guard piping around hydrogen lines is not required; however, consideration shall be given to its use for such purposes as protection from heavy traffic loads, leak detection and monitoring, or isolation of the potential hazard from nearby equipment, etc. All hydrogen piping should be grounded and have electrical continuity.

Excess flow valves should be installed in the hydrogen line at appropriate locations to restrict flow out of a broken line. Excess flow protection shall be designed to ensure that a line break will not result in an unacceptable hazard to personnel or equipment (BTP CMEB 9.5-1). The design features for mitigating the consequences of a leak or line break must perform their intended design function with or without normal ventilation.

Individual pump injection lines shall contain a check valve to prevent feedwater from entering the hydrogen line and to protect upstream hydrogen gas components. Automatic isolation valves should be provided in each injection line to prevent hydrogen injection into an inactive pump.

Purge connections shall be provided to allow the hydrogen piping to be completely purged of air before hydrogen is introduced into the line. Nitrogen or another inert gas shall be used as the purge gas. Gases shall be purged to safe locations, eithe tly or through intervening flow paths, such that personnel or explosive hazards are not encountered and undesirable quantities of gas are not injected into the reactor.

Area hydrogen concentration monitors are an acceptable way to ensure that hydrogen concentration is maintained below the flammable limit. If used, such monitors should be located at high points where hydrogen might collect and/or above use points that constitute potential leaks. Good engineering practice for locating hydrogen detector heads is to take into consideration the positive buoyancy of gaseous hydrogen. Detector heads shall be located so that the monitors shall be capable of detecting hydrogen leaks with or without normal ventilation. Each utility shall evaluate its particular system design and identify specific points where hydrogen concentration monitors should be installed. Examples of such points include flanged in-line devices (such as calibration spool pieces associated with mass flowmeters), outlets of purge/vent paths, or the items discussed in the following paragraph. Sleeves or guard pipes can be used as an alternative method to mitigate the consequences of a line break.

A hydrogen addition system will increase the hydrogen concentration in the feedwater, reactor, steam lines and main condenser. Each of these systems shall be reviewed for possible detrimental effects. A discussion of possible concerns is presented below.

1. Main Condenser

   The main condenser presently handles combustible gases. The hydrogen addition system does not significantly change the concentration or volume of noncondensables. Therefore, it is not anticipated that hydrogen addition will affect operation of the main condenser.

2. Off-Gas System

   Oxygen shall be added into the off-gas system to recombine with the hydrogen flow thus limiting the extent of the system handling hydrogen rich mixtures and reducing volumetric flow rates. The net effect will probably be a revised heat input into the recombined off-gas. The capability of the off-gas system to handle this revised heat load must be evaluated to ensure that temperature limits are not exceeded. Considerations in the design of the off-gas oxygen injection system should include loss of oxygen and runaway oxygen injection.

3. Steam Piping and Torus

   Hydrogen water chemistry may slightly increase the rate of hydrogen leakage into the torus via the safety relief valves. However, the rate of oxygen leakage will be decreased. Thus, the possibility of forming a combustible mixture is not significantly increased when compared to non-HWC operation.

Section 3

SUPPLY FACILITIES

3.1 GASEOUS HYDROGEN

3.1.1 System Overview

Hydrogen gas can be supplied from either permanent high-pressure vessels or from transportable tube trailers. For the permanent storage system, gaseous hydrogen is stored in seamless ASME code vessels at pressures up to 2,400 psig and ambient temperatures. Transportable vessels are designed to DOT standards and store hydrogen at pressures up to 2650 psig at ambient temperatures. With either storage design, the gas is routed through a pressure control station which maintains a constant hydrogen supply pressure. In any event, the gaseous hydrogen system shall be provided by a supplier who has extensive experience in the design, operation and maintenance of associated storage and supply systems. Gaseous hydrogen shall be provided per CGA G-5 and G-5.3.

3.1.2 Specific Equipment Description

3.1.2.1 Hydrogen Storage Vessels. The hydrogen storage bank shall be composed of ASME Code gas storage vessels. Each tube shall be constructed as a seamless vessel with swagged ends. Specific tube design shall be based on ASME Unfired Pressure Vessel Code, Section VIII, Division 1, including Appendix XIV-70.

The tube bank shall be supported to prevent movement in the event of line failure and each tube shall be equipped with a close-coupled shutoff valve. As an alternative, one safety valve per bank of tubes can be used, provided the safety valve is sized to handle the maximum relief from all tubes tied into the valve. Each bank shall be equipped with a thermometer and a pressure gauge, as is necessary for proper filling.

3.1.2.2 Transportable Hydrogen Storage Vessel. Transportable hydrogen vessels shall be constructed, tested, and retested (every 5 years), in accordance with DOT specifications 3A, 3AA, 3AX, or 3AAX. All valving and instrumentation shall be identical to section 3.1.2.1.

3.1.2.3 Pressure Reducing Station. The pressure control station shall be of a manifold design. The manifold shall have two (2) full-flow parallel pressure reducing regulators. The discharge pressure range of these regulators shall be adjustable to satisfy plant hydrogen injection requirements. Pressure gauges shall be provided upstream and downstream of the regulators. Sufficient hand valves shall be provided to ensure complete operational flexibility.

An excess flow check valve shall be installed in the manifold immediately downstream of the regulators to limit the flow rate in the event of a line break. The stop-flow set point shall be determined by each plant and should be set between the maximum plant flow requirements and the full C_v of the flow control valves. Additional guidance on excess flow protection is provided in section 2.3.1.3.

3.1.2.4 Tube Trailer Discharge Stanchion. A tube trailer discharge stanchion shall be provided for gaseous product unloading. The stanchion shall consist of a flexible pigtail, shutoff valve, check valve, bleed valve, and necessary piping. Filling apparatus shall be separated from other equipment for safety and convenience, and protected with walls or barriers to prevent vehicular collision.

A tube trailer grounding assembly shall be provided for each discharge stanchion to ground the tube trailer before the discharge of hydrogen begins.

3.1.2.5 Interconnecting Pipeline. All equipment and interconnecting piping supplied with this system shall be installed in compliance with the following standards:

- American National Standards Institute (ANSI) B31.1, Power Piping, or B31.3, Chemical Plant and Petroleum Refinery Piping.
- National Fire Protection Association (NFPA) 70, National Electrical Code.
- NFPA-50A, Bulk Hydrogen Systems.
- All applicable local and national codes.

There are several suitable field installation techniques which are based on industrial experience. The following are guidelines which may be used for field connections:

- Copper-to-Copper, Brass-to-Brass, and Copper-to-Brass Socket Braze Joints.

--Silver Alloy	45% Ag, 15% Cu, 16% Zn, 24% Cd., ASTM B260-69T and AWS A5.8-69T, BAg-1 Melting Range-Solidus-607.2°C Liquidus-618.3°C
--Flux	Working Range 593.3°C to 871.1°C

- Copper, Brass, Carbon Steel, and Stainless Steel N.P.T. Threaded Joints.

--TEFLON* Tape**	SCOTCH*** Number 48 Tape** or equal. -195.5°C to +204.4°C, 0 to 3,000 psig. Wrapped in direction of threads.

- Flange Joints (On all Materials).

--Ring Gasket Material, Low Pressure (720 psig maximum)	Precut T.F.E. impregnated asbestos, 1/16 inch thickness. Garlock 900 or equal. -195.5°C to +168.3°C, 0 to 900 psig.
--Ring Gasket Material, High Pressure	FLEXITALLIC**** Type. Material to be 0.175 inch thick 304 stainless steel with TEFLON filler and 0.125 inch carbon steel guide ring.

*TEFLON is a trademark of E. I. duPont de Nemours & Co., Wilmington, DE 19898.

**If tape is used, electrical continuity/grounding of each piping section should be confirmed.

***SCOTCH is a trademark of 3M Company, St. Paul, MN 55101.

****FLEXITALLIC is a trademark of Flexitallic Gasket Co., Bellmawr, NJ 08031.

--Antiseize Compound	For flange face, nut, and bolt lubrication. Halocarbon 25-5S grease or equal. -195.5°C to +176.6°C, 0 to 3,000 psig. DO NOT USE ON ALUMINUM, MAGNESIUM, OR THEIR ALLOYS UNDER CONDITIONS OF HIGH TORQUE OR SHEAR.

- Carbon Steel, Stainless Steel, and Aluminum Alloys Socket and Butt Welds.

--Welding Procedure	Gas Metal Arc Welding (GMAW), Gas Tungsten Arc Welding (GTAW), Shielded Metal Arc Welding (SMAW), or Plasma Arc Welding (PAW); with appropriate filler material and shielding gas. Proper surface and joint preparation (in regard to cleaning and clearances) should be exercised.

3.1.2.6 Component Cleaning. All components that contact hydrogen must be free of moisture, loose rust, scale, slag, and weld spatter; they must be essentially free of organic matter, such as oil, grease, crayon, paint, etc. To meet these objectives, system components shall be cleaned in accordance with standard industrial practices, as recommended by the gas supplier, prior to and following system fabrication.

3.2 LIQUID HYDROGEN

3.2.1 System Overview

Liquid hydrogen is stored in a vacuum-jacketed vessel at pressures up to 150 psig and temperatures up to -403°F (saturated). Based on data relating hydrogen injection pressures to BWR plant power levels, hydrogen supply from a liquid source can be provided directly from a tank or pumped into supplemental gaseous storage. Gaseous storage requirements are identified in section 3.1. The required supply pressure shall be based on pressure requirements at the point of hydrogen injection and line losses from the hydrogen supply system to the injection point.

Feedwater pressure requirements and line losses must not exceed 120 psig if hydrogen is to be supplied directly from a liquid tank.

In any event, the liquid hydrogen system shall be provided by a supplier who has extensive experience in the the design, operation and maintenance of associated storage and supply systems, such as cryogenic pumping. Liquid hydrogen shall be provided in accordance with CGA G-5 and G-5.3.

3.2.2 Specific Equipment Description

3.2.2.1 Cryogenic Tank. Tanks for liquid hydrogen service are available with capacities between 1,500 gallons and 20,000 gallons. An "inner vessel" or "liquid container" is supported within an "outer vessel" or "vacuum jacket," with the space between filled with insulation and evacuated. Necessary piping connects from inside of the inner vessel to outside of the vacuum jacket. Gages and valves to indicate the control of hydrogen in the vessel are mounted outside of the vacuum jacket. Legs or saddles to support the whole assembly are welded to the outside of the vacuum jacket.

Inner vessels are designed, fabricated, tested, and stamped in accordance with Section VIII, Division 1 of the ASME Code for Unfired Pressure Vessels. Materials suitable for liquid hydrogen service must have good ductility properties at temperatures of -422°F per CGA G-5. The cryogenic operating temperatures of these vessels preclude material degrading mechanisms such as corrosion or hydrogen embrittlement. The constant operating vessel pressures assure that flaw growth due to cyclic stress loading will not occur. The inner vessel is subject to a required pressure test which insures that no flaws exist that could cause a failure at or below the set pressure of the vessel's redundant relief devices. In addition to ASME Code inspection requirements, 100% radiography of the inner vessel longitudinal welds shall be completed. The tank outer vessel shall be constructed of carbon steel and shall not require ASME certification.

Insulation between inner and outer vessels shall be either perlite, aluminized mylar, or suitable equal. Fibrous or blanket insulation, such as bonded glass fibers or rock wool, shall not be used because of the potential for liquid-saturated missiles which would occur only as a result of vessel failure. The annular space should be evacuated to a high vacuum of 50 microns or less.

Tank control piping and valving should be installed in accordance with ANSI B31.1 or B31.3. All piping shall be either wrought copper or stainless steel. The following tank piping subsystems shall be provided:

- Fill circuit, constructed with top and bottom lines so that the vessel can be filled without affecting continuous hydrogen supply.
- Pressure-build circuit, to keep tank pressures at operational levels.
- Vacuum-jacketed liquid fill and pump circuits, where applicable.

3.2.2.2 Overpressure Protection System. Safety considerations for the tank shall be satisfied by dual full-flow safety valves and emergency backup rupture discs. The primary relief system shall consist of two sets of a minimum of one (1) rupture disk and safety valve piped into separate "legs." Relief devices shall be connected in parallel with other relief devices. The system shall be coupled by a 3-way diverter valve or tie bar interlock so that one leg is opened when the other is closed. With this arrangement, a minimum of one safety valve and one rupture disk will be available at all times. The dual primary relief systems with 100% standby redundancy allows maintenance and testing to be performed without sacrificing the level of protection from overpressure.

The primary relief system shall comply with the provisions of the American Society of Mechanical Engineers (ASME) Pressure Vessel Codes and the Compressed Gas Association (CGA) Standards.

The tank shall also be supplied with a secondary relief system not required by the ASME Codes. This system shall be totally separate from the primary relief system. It shall consist of a locked open valve, a rupture disk, and a secondary vent stack. This rupture disk shall be designed to burst at 1.33 times maximum allowable working pressure (MAWP).

Supply system piping that may contain liquid and can be isolatable from the tank relief valves shall be protected with thermal relief valves. All outlet connections from the safety relief valves, rupture devices, bleed valves, and the fill line purge connections shall be piped to an overhead vent stack, per CGA G-5, Section 7.3.7.

Two relief devices shall be installed in the tank's outer vessel to relieve any excessive pressure buildup in the annular space.

Hydrogen tanks and delivery vehicles shall be grounded per CGA P-12, Sections 5.4.5 and 5.7.1.1. The storage system shall be protected from the effects of lightning per NFPA 78, Chapter 6.

Excess flow protection shall be added to the tank's liquid piping wherever a line break would release a sufficient amount of hydrogen to threaten safety-related structures. An acceptable methodology is identified in section 4.2.2, "Pipe Breaks."

3.2.2.3 Instrumentation. The tank shall be supplied with a pressure gauge, a liquid level gauge, and a vacuum readout connection. These gauges are sufficient for normal monitoring of the tank condition. Instrumentation for remote monitoring, such as high/low-pressure switches, pressure and level transmitters may be added. A listing of supply system instrumentation and control is identified in section 2.4.

3.2.2.4 Liquid Hydrogen Pump and Controls. The liquid hydrogen pump shall be of proven design to provide continuous hydrogen supply in unattended, automatic operation. The following items comprise the more important system controls.

3.2.2.4.1 Positive isolation valve. A positive isolation valve shall be used to control the liquid feed into the pumping system per NFPA 50B. The valve shall be a failed-closed, pneumatically operated valve. The valve shall only be open during pump operation, shall close in any fault mode, and shall be able to be remotely overridden in case of emergency.

3.2.2.4.2 System overpressure shutdown. Although the system is protected by safety relief valves and rupture discs, system overpressure shall be avoided by shutting down the pumps at high pressure.

3.2.2.4.3 Temperature indicating switch. A temperature switch shall continuously monitor the downstream gas line for low temperature and shall trip the liquid pump to protect downstream equipment from low temperatures.

3.2.2.4.4 Pump operation. Pump operation shall be continuously and automatically monitored. Operation which results in pump cavitation, high temperature at the pump discharge, or low temperature downstream of the vaporizer shall cause the pump to be shut down by the remote control panel. The fault shall be indicated on the remote control panel by an audible alarm and light indication.

3.2.2.4.5 Purging of controls. All electrical components in hydrogen service should be designed in accordance with NFPA 70. Only nitrogen or another inert gas shall be used for purging pump motors, control panels and valves.

3.2.2.5 Interface with Gaseous System. Liquid hydrogen pump systems typically require a gaseous storage system as a surge or back-up to plant hydrogen supply. These storage systems shall be designed in accordance with section 3.1, Gaseous Hydrogen. Whenever a gaseous backup is used in conjunction with a liquid hydrogen system, switchover controls shall be provided.

3.2.2.6 Vaporization. Vaporization of the liquid hydrogen shall be achieved by the use of ambient air vaporizers. Vaporizer design, installation and operation shall take guidance from NFPA 50A and 50B.

The vaporizer should feature a star fin design and aluminum alloy construction. For a combined liquid and gaseous storage system, the vaporizers used should have a design pressure consistent with plant injection pressure requirements. The units may be piped in parallel such that each unit can operate independently. Parallel vaporizer assemblies shall be sized for the peak hydrogen flow required for each plant and shall provide for periodic intervals for defrosting, as appropriate. Other atmospheric vaporization systems may be utilized if their capacity is demonstrated to be adequate for the plant flow and ambient conditions.

For a pumped liquid only storage system, the vaporizer must withstand maximum pressures generated from the cryogenic pump. These vaporizers shall be equipped with stainless steel lining designed to 3500 psig.

3.3 ELECTROLYTIC

3.3.1 System Overview

The disassociation of water by electrolysis is an acceptable method of obtaining the gases needed for hydrogen water chemistry. This can be done on site and the gases can conveniently be generated at the rate used. The electrolytic gas generator should be proven equipment, the same as used in other industrial

Section 4

SAFETY CONSIDERATIONS

4.1 GASEOUS HYDROGEN

4.1.1 Site Characteristics of Gaseous and Liquid Hydrogen

4.1.1.1 Overview. Review of the following site characteristics shall be conducted by each BWR facility in locating the gaseous and/or liquid hydrogen supply systems:

1. Location of supply system in proximity to exposures as addressed in NFPA 50A and 50B.
2. Route of hydrogen delivery on site.
3. Location of supply system in proximity to safety-related equipment.

4.1.1.2 Specific Considerations.

4.1.1.2.1 Fire protection. The area selected for hydrogen system siting shall meet or exceed all requirements for protection of personnel and equipment as addressed in NFPA 50A and 50B, gaseous and liquified hydrogen systems, respectively. Each standard identifies the maximum quantity of hydrogen storage permitted and the minimum distance from hydrogen systems to a number of exposures.

The need for additional fire protection for other than the hydrogen facility shall be determined by an analysis of local conditions of hazards on-site, exposure to other properties, water supplies, and the probable effectiveness of plant fire brigades in accordance with NFPA 50A and 50B.

4.1.1.2.2 Security. All hydrogen storage system installations shall be completely fenced, even when located within the owner-controlled area. Lighting shall be installed to facilitate night surveillance.

4.1.1.2.3 Route of hydrogen delivery on site. Each plant should determine the route to be taken by hydrogen delivery trucks through on-site and off-site areas. In order to protect the hydrogen storage area from any vehicular accidents, truck barriers shall be installed around the perimeter of the system installation.

Within the plant security area, all deliveries shall be controlled per the requirements of 10 CFR 73.55.

4.1.1.2.4 Location of storage system to safety-related structures. Each plant shall determine that the location of the hydrogen storage system is acceptable relative to safety-related structures and equipment considering the hazards described in sections 4.1.2, 4.1.3, 4.2.1 and 4.2.2.

4.1.2 Gaseous Storage Vessel Failure

Gaseous storage vessels in the scope of this report are the commercially available, seamless, swagged-ended vessels that are commonly referred to as "hydril tubes." This section addresses the non-mechanistic rupture failure of single vessels and the separation distances required to avoid damage to safety-related equipment. Simultaneous failure of multiple vessels is not addressed because the inherent strength of the vessel makes them unsusceptible to failure from outside forces. These vessels shall be capable of withstanding tornado missiles (NUREG-0800) and site specific seismic loading due to horizontal and vertical accelerations acting simultaneously.

These features eliminate common cause vessel failures so that the maximum postulated instantaneous release is the fully pressurized contents of the largest single vessel. The potential consequences of such a release, a fireball or an explosion, are addressed in order.

4.1.2.1 Fireball. The thermal flux versus distance from the fireball center are shown on figure 4-1 for the two most common vessel sizes. These fluxes and durations will not adversely affect safety-related structures. However, each utility shall review any unique site characteristics to assure all safety-related equipment will function in the event of a fireball.

4.1.2.2 Explosion. When a gaseous storage vessels ruptures, the expansion of the high-pressure gas results in rapid turbulent mixing with the surrounding air. In

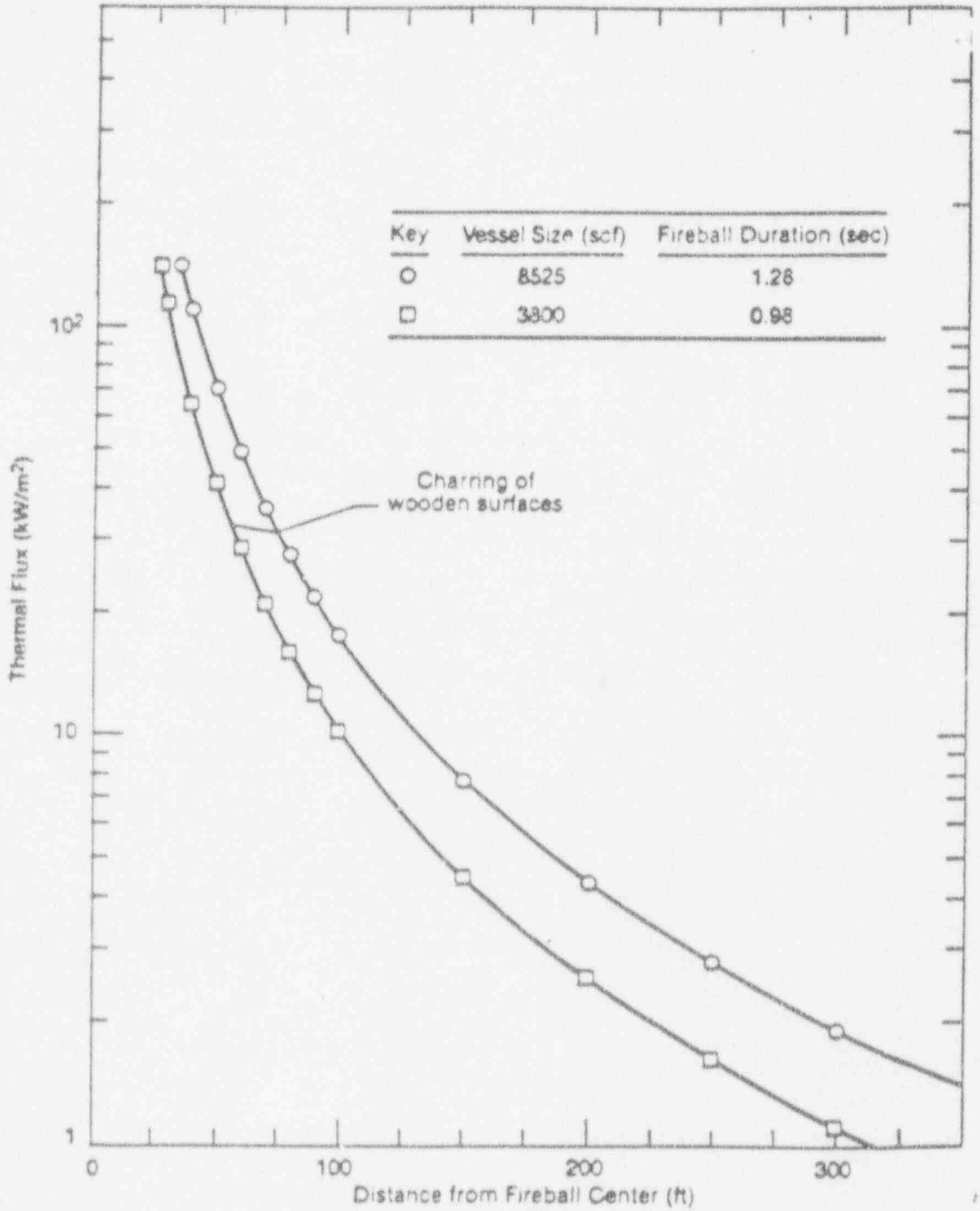

Figure 4-1. Thermal flux vs. distance from fireball center for gaseous hydrogen storage system.

the case of gaseous hydrogen, the release will go through the detonation limits of 18.3 - 59% before the wind can translate the mixture. Consequently, any explosion blastwaves will originate at the vessel rupture site. For this report, it is conservatively assumed that 100% of the vessel contents will contribute to the blastwave and that the TNT-hydrogen equivalence is 20% on an energy basis (520% on a mass basis). This translates to 27.1 lbs. of TNT per 1000 standard cubic feet (SCF) of gaseous hydrogen. Using this conversion factor and U.S. Army Technical Manual TM5-1300, blast overpressures and impulses can be calculated as functions of distance from the vessel location. These blast parameters could then be compared to the dynamic strength of safety-related structures.

An evaluation entitled "Separation Distances Recommended for Hydrogen Storage to Prevent Damage to Nuclear Power Plant Structures From Hydrogen Explosion" was performed for EPRI by R. P. Kennedy. This evaluation, which is included as appendix B of these guidelines, recommends separation distances based on quantities of stored hydrogen and building design factors. The recommendations are provided in the form of step-by-step procedures, with subsequent steps requiring additional work but resulting in reduced distances from the previous step. The procedure to determine acceptable separation distances is outlined below.

1. For any reinforced concrete or masonry walls at least 8-inches thick, the upper curve on figure 4-2 provides conservative separation distances as a function of vessel size. If this is acceptable, then no further work is needed. Otherwise, proceed to step 2.

2. For reinforced concrete walls at least 18-inches thick, with known static strength and percent tensile rebar, eq. 7 in appendix B can be used to determine required separation distances. The two lower curves on figure 4-2 are representative examples of design parameters for walls of nuclear power plants. Walls with different parameters should be analyzed using the methods in appendix B, pages 10 through 13. If this is acceptable, then no further work is needed. Otherwise, proceed to step 3.

3. For separation distances closer than allowed by the above 1 and 2, perform a dynamic blast capacity analysis in accordance with NUREG/CR-2462 (1).

For all storage locations, the vessel(s) and the foundation(s) shall be designed to remain in place for both design-basis tornado characteristics and site-specific flood conditions.

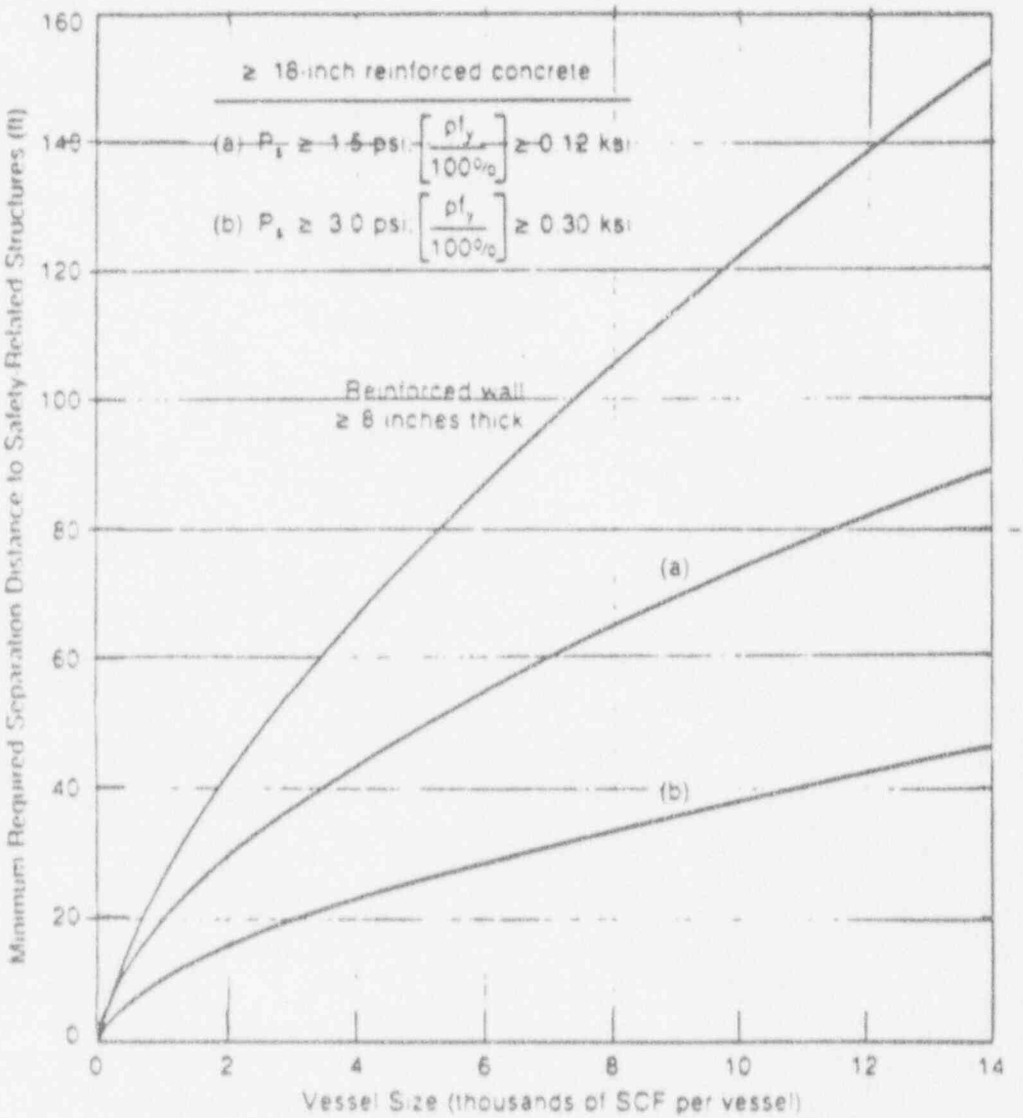

Figure 4-2 Minimum required separation distances to safety-related structures versus vessel size for gaseous hydrogen storage system.

4.1.3 Gaseous Pipe Breaks

This section addresses the requirements for hydrogen piping systems attached to gaseous storage vessels up to the point where excess flow protection is provided. The criteria for acceptable siting for the event of a pipe break are:

1. Dilution of resultant release below the lower flammability limit of 4% before reaching air pathways into safety-related structures.

2. Minimum separation distances for the blast damage criteria outlined in section 4.1.2.

It is conservatively assumed that all releases occur while the storage vessel is at 2,450 psig. This is the maximum allowable working pressure of the majority of commercially available vessels.

Gaseous releases at elevated pressures result in supersonic jet velocities and a dispersion process that is momentum-dominated. Under these conditions, the Gaussian dispersion model unrealistically overestimates the amount of hydrogen in the explosive region and the distance to the lower flammable region. Therefore, these properties of gaseous releases were calculated using a jet dispersion model described in reference (2).

The results of this modeling are shown in figure 4-3 as minimum separation distances versus inside diameter of the pipe. The upper curve is the maximum distance to the lower flammability limit of 4% hydrogen. Each utility shall determine that the location of air pathways into safety-related structures exceeds this minimum separation distance or show that other criteria should be applied to a specific case. An example of such an exception would be if the air intakes have automatic shutters controlled by hydrogen analyzers thus preventing the ingestion of a flammable mixture.

The lower curve on figure 4-3 is the minimum required distance to safety-related structures with greater than or equal to an 8-inch thick reinforced mansonry or concrete wall. This distance includes the drift distance of an unignited, fully developed gaseous jet plus the blast distance for the maximum amount of hydrogen in the detonable region. It conservatively assumes that the pipe break is oriented directly toward the safety-related structures. Each utility shall determine compliance with this minimum separation distance or demonstrate that other criteria should be applied.

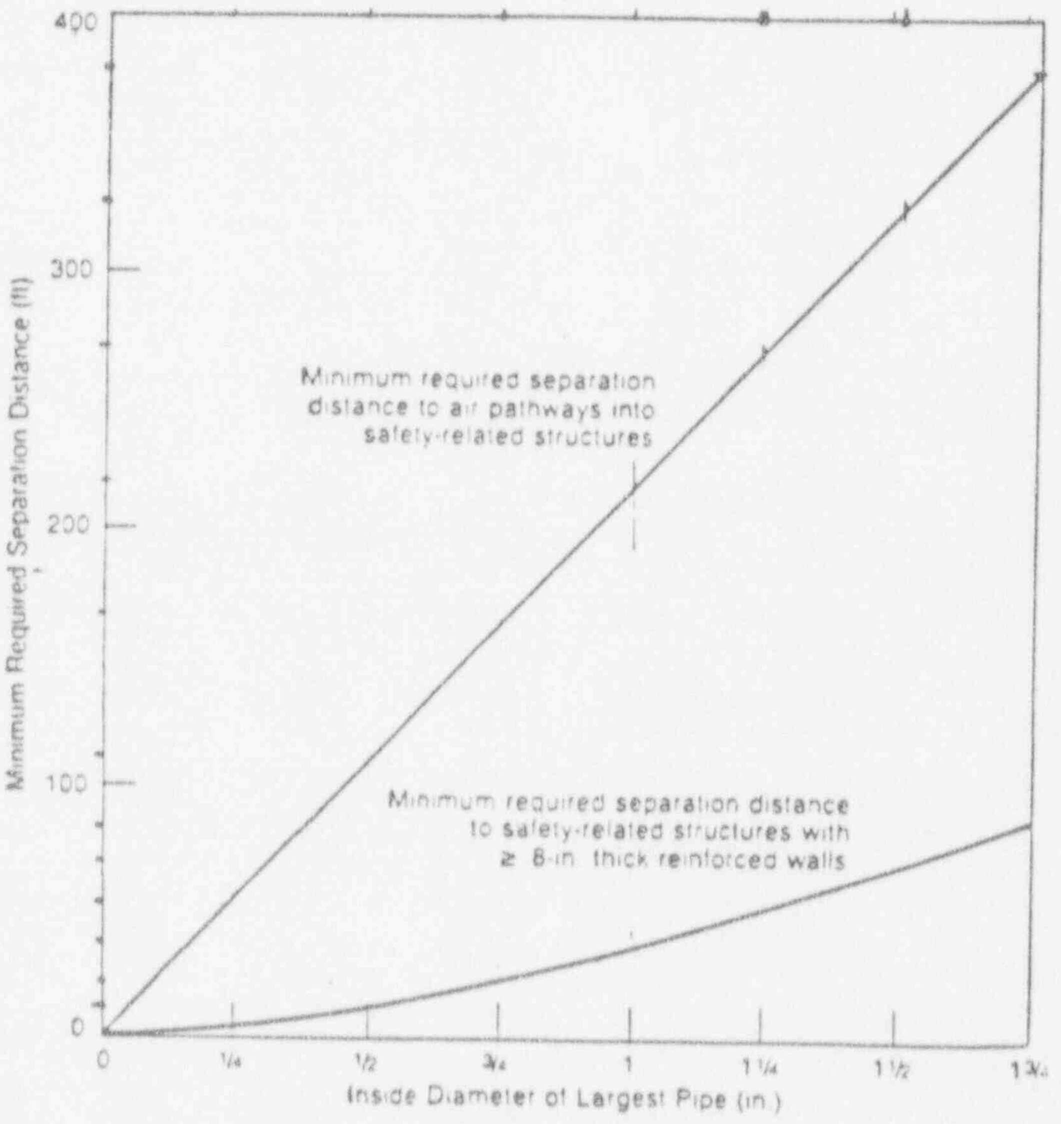

Figure 4-3 Minimum required separation distance versus ID of pipe for releases from 2450 psig gaseous hydrogen storage systems

4.2 LIQUID HYDROGEN

4.2.1 Storage Vessel Failure

For this report, storage vessel failure is defined as a large breach resulting in the rapid emptying of the entire contents of liquid hydrogen. It is assumed that the tank is full at the time of failure and that the entire spill vaporizes instantaneously. The following enumerates potential causes of vessel failure and the required design features that mitigate or alleviate these potentials.

- Seismic

  The tank and its foundation shall be designed to meet the seismic criterion for critical structures and equipment at the plant site (i.e., design basis earthquake). It is preferable to seismically support all liquid hydrogen piping. If this is not possible, the liquid hydrogen piping shall be seismically supported up to and including excess flow protection devices. The specific liquid hydrogen tank and piping design at each installation shall meet these requirements.

- Tornado and Tornado Missiles

  The tank and its foundation shall be designed to withstand the "design basis tornado characteristics" as outlined in Regulatory Guide 1.76. As a minimum, the tank shall remain in place so that any liquid spillage will originate from the tank location. The specific tank and foundation design at each installation shall meet these requirements.

  Design basis tornado-generated missiles are capable of breaching all known commercially available liquid hydrogen storage vessels. Therefore, tornado missiles are a potential cause of "storage vessel failure."

- Aircraft

  A large aircraft crashing directly into the storage area is capable of breaching all known commercially available liquid hydrogen storage vessels. Therefore, aircraft crash is a potential cause of "storage vessel failure."

- Fire

  The overpressure protection system shall be sized to accommodate the worst-case vaporization rate caused by a hydrocarbon fire engulfing the outer shell with loss of vacuum and hydrogen in the annulus of the double-wall storage tank (as per Compressed Gas Association 5.3 and ASME Section VIII requirements).

Flood

The following flood conditions could result in vessel failure:

--High water reaches the top of the vent stack for the overpressure protection system.

--High flood velocities dislodge the tank.

Under either condition, water could enter the vent system and defeat the overpressure protection system. Therefore, the tank shall be located such that maximum flood heights cannot exceed the vent stack elevation and such that potential flood velocities cannot damage the vent stack or dislodge the tank.

- Vehicle Impact

  The storage vessel shall be protected from the impact of the largest vehicle used on-site by a barricade capable of stopping such a vehicle.

- Vessel Structural Failure

  The storage vessel shall be designed, constructed, inspected and operated to assure an extremely low likelihood of tank structural failure during its tenure on site. A vessel designed in accordance with this document complies with this low-probability requirement.

4.2.1.1 Fireball. For the two potential causes of "storage vessel failure," tornado missiles and aircraft impact, a fireball at the tank location is the expected result. The major reasons for this is the high ignitability of hydrogen and the density of ignition sources in the aftermath of these causal events. An aircraft impact or a design basis tornado and the associated missiles will also provide numerous sources of ignition from downed power lines, damaged transformers, and switchgears, etc. Details of these considerations are given in the report for the Dresden plant (2).

The thermal flux versus distance from the fireball center (tank location) is shown on figure 4-4 for the range of commercially available tank sizes. The durations of the various fireball sizes are also given. These fluxes and durations will not adversely affect equipment or personnel enclosed in concrete/steel safety-related structures. However, each utility shall review any unique site characteristics to assure all safety-related equipment will function in the event of a fireball.

4.2.1.2 Explosion at Tank Site. Although an explosion is not expected, safety-related structures and equipment shall be verified to be capable of withstanding a detonation occurring at the site of the tank installation. For the instantaneous

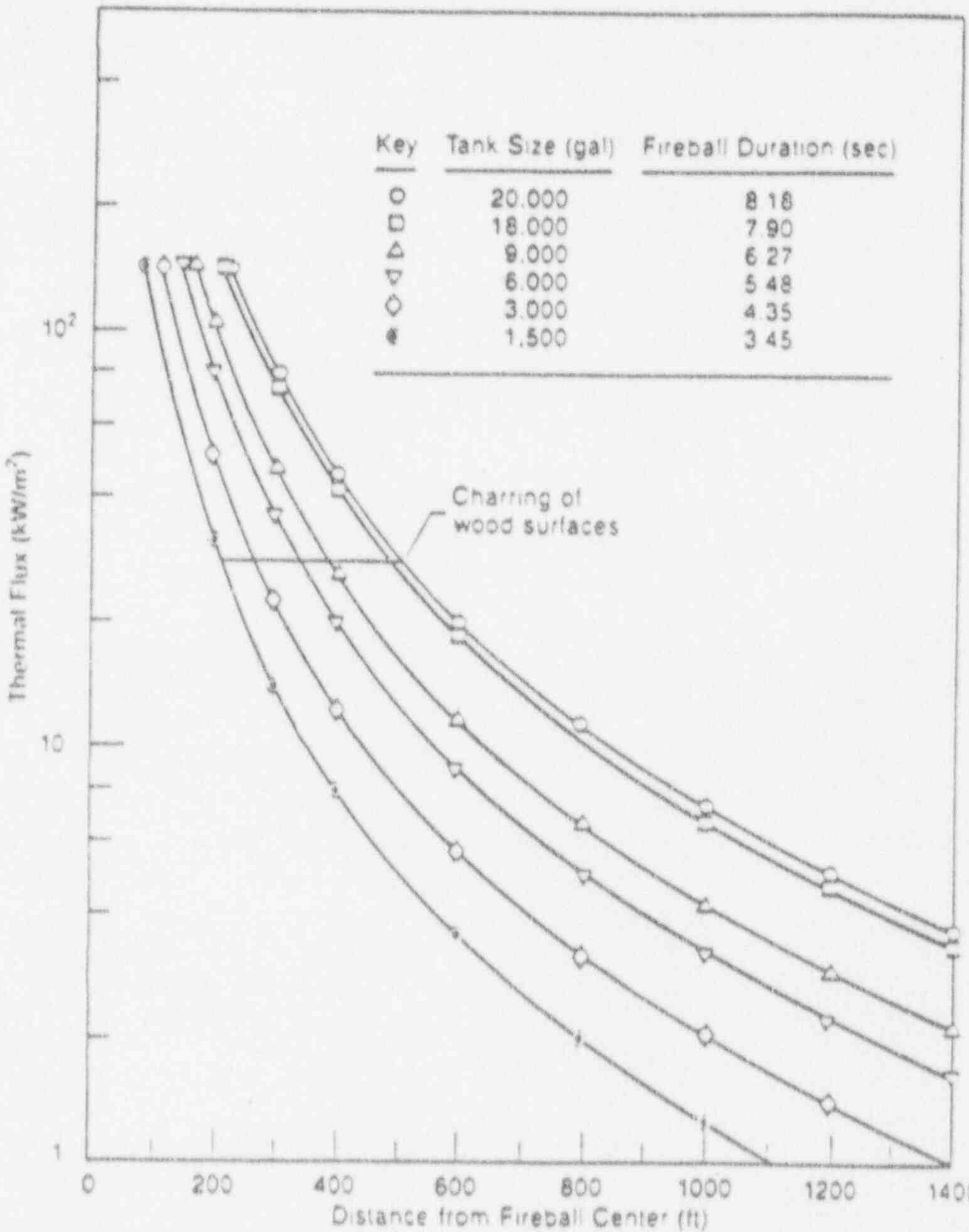

Figure 4-4. Thermal flux versus distance from fireball center for liquid hydrogen storage system

release of the entire tank contents, the following were used to determine blast parameters for an explosion at the tank site:

1. Gaussian F weather stability
2. Detonation limits of hydrogen, 18.3-59%
3. TNT - hydrogen equivalent of 20% on an energy basis (520% on a mass basis)

NUREG/CR-2726 reports that detonations have been observed for hydrogen concentrations as low as 13.8% when ignited in a long, large-diameter tube. The explosive yield or TNT equivalence of such threshold concentration reactions is extremely low because most of the combustion energy is expended in the transition to detonation. This is essentially the reason why it represents the lower detonation limit; any less concentration will give a zero detonation yield. This also points out that both hydrogen concentration and explosive yield affect the total equivalent mass of TNT for a given release.

Regulatory Guide 1.91 models the blast effects from transportation accidents by assuming 100% of the cargo detonates at a TNT mass equivalence of 240% (one pound of cargo equals 2.4 pounds of TNT). The analysis described in this report modeled large spills of hydrogen by calculating the amount of release that is between 18.3 and 59% (~46% of the vessel contents) and assuming that it detonates at a TNT mass equivalence of 520%. The resulting TNT equivalence for this method is one pound of vessel contents equals 2.4 pounds of TNT, an identical result to that obtained with the NRC method.

The above results in an equivalence of 1.37 lbs of TNT per gallon of tank size. Using this conversion factor and U.S. Army Technical Manual TM5-1300 and the damage criteria outlined in appendix B, required separation distances have been determined as a function of tank size. The results are shown on figure 4-5 for the design parameters of the three building types described in section 4.1.2.2. For buildings with other design parameters, the methods in appendix B or in NUREG/CR-2462 (1) may be used to determine separation distances. Each utility shall use these methods for determining the minimum required separation distances from the storage tank to safety-related structures or equipment for the event of an explosion at the tank site.

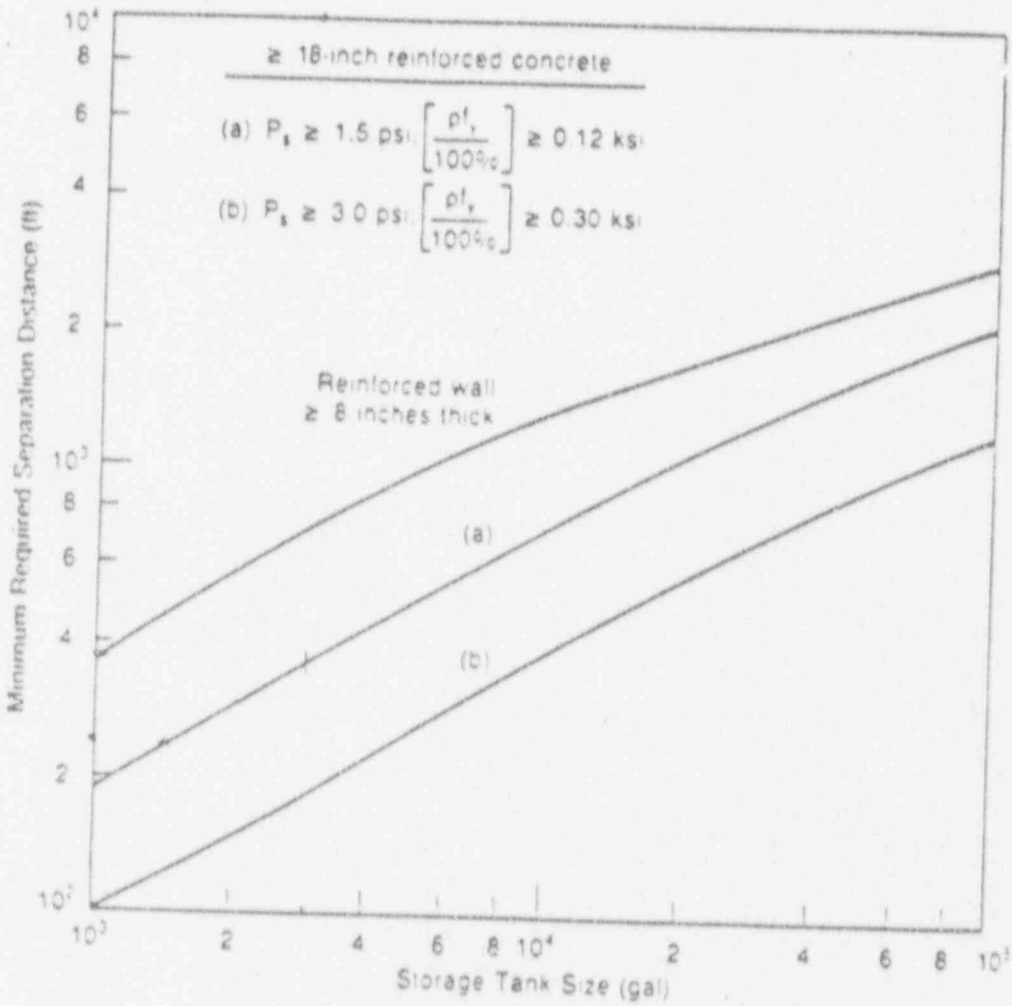

Figure 4-5 Minimum required separation distance versus liquid hydrogen storage tank size for instantaneous release of entire tank contents and explosion at tank site F weather stability

4.2.2. Pipe Breaks

This section addresses the requirements for gaseous and liquid hydrogen piping systems attached to the storage vessel up to the point where excess flow protection is provided. The criteria for acceptable siting for the event of a pipe break are the same as outlined in section 4.1.3. It is conservatively assumed that all releases occur while the storage vessel is at 150 psig (the maximum allowable working pressure of the majority of commercially available tanks).

4.2.2.1. Gaseous Piping. The same dispersion model for momentum-dominated jets discussed in section 4.1.3 applies to gaseous releases from liquid storage tank piping with the appropriate release conditions for saturated vapors. The results of this modeling are shown in figure 4-6 as minimum separation distances versus hole size or inside diameter of piping not protected with excess flow devices. The upper curve is the maximum drift distance to the lower flammability limit and is the minimum required separation distance to air pathways into safety-related structures. The three lower curves are required separation distances for the representative types of safety-related structures. These distances are the sum of both the drift and blast distances. Structures with other parameters can be analyzed using the methods in appendix B or in NUREG/CR-2462 (1). Each utility shall determine that the storage vessel piping and location meet these minimum requirements or show that less stringent criteria should be applied to a specific case. An example of such a suitable exception would be if the air intakes are provided with automatic shutters controlled by hydrogen analyzers to prevent the ingestion of a flammable mixture.

4.2.2.2 Liquid Piping. The vapor cloud formed by the flashing and rapid vaporization of a liquid release is nearly neutrally buoyant and has little momentum associated with its formation. For these conditions, a Gaussian dispersion model is employed using the following conservative assumptions:

1. Instantaneous vaporization of release
2. F weather stability
3. 1 m/s wind speed
4. Wind direction towards safety-related area

No credit is to be taken for site-specific wind direction or speed characteristics since it is assumed that pipe breaks can occur during the worst-case weather and wind conditions.

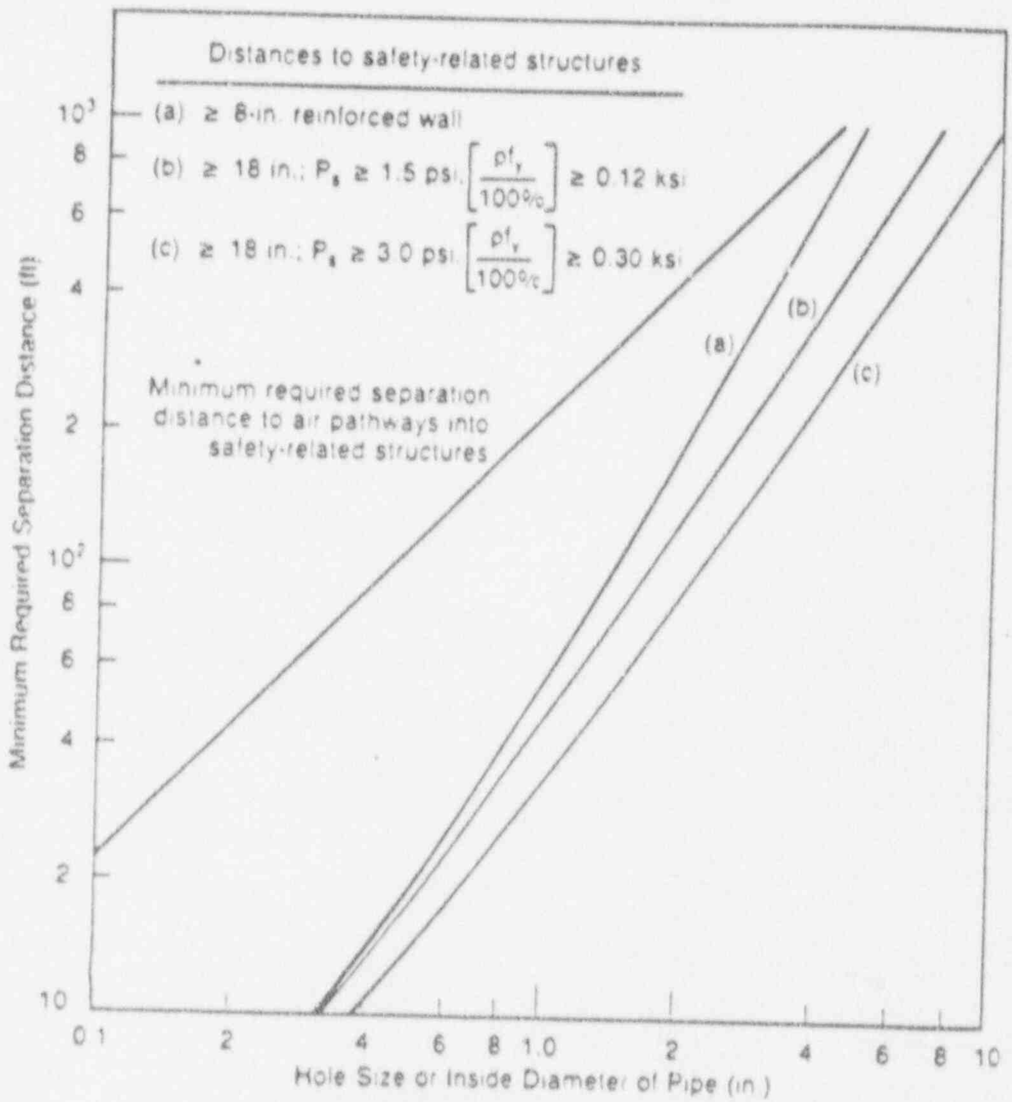

Figure 4-6 Minimum required separation distance versus hole size and ID of pipe for gaseous releases from 150 psig liquid hydrogen storage tank

The minimum required separation distances for liquid hydrogen pipe breaks, using the above assumptions, are given on figure 4-7 as a function of discharge rate and hole size The upper curve is the drift distance to the lower flammability limit for a fully developed could with F stability and 1 m/s windspeed. This defines the minimum required separation distance to air pathways into safety-related structures. The three lower curves define the minimum required separation distances to the representative safety-related structures. These curves include the drift distance to the center of the detonable cloud and the blast distance for the amount of hydrogen in the detonable region. For other structure types, appendix B or NUREG/CR-2462 (1) may be used to determine blast distances. These distances shall be applied to all liquid piping, including those from any pump discharges, that are not seismically supported or protected by excess flow devices.

4.3 ELECTROLYTIC

4.3.1 General

The electrolytic supply option need not constitute storage of hazardous materials on-site if it operates at approximately atmospheric pressure and involves the storage of no more than 2500 scf of hydrogen and 250 scf of oxygen. If these limits are met, and the system is designed as described in section 3.3, it need only be analyzed as described below. Other system designs have not yet been considered. Compressed gases utilized in conjunction with electrolytic systems shall be in accordance with sections 3.1 and 4.1.

Events important to industrial safety (abnormal transients, accidents and external events) must be evaluated to identify those which could result in any of the following conditions:

1. Hydrogen accumulation to a combustible mixture in an enclosed space.
2. Air or oxygen mixing with hydrogen within electrolytic system components.
3. Hydrogen fires.

When the potential exists for the above undesired conditions to occur, appropriate mitigating features shall be incorporated in the design or operation of the system or the consequences with respect to plant and personnel safety shall be evaluated by the owner.

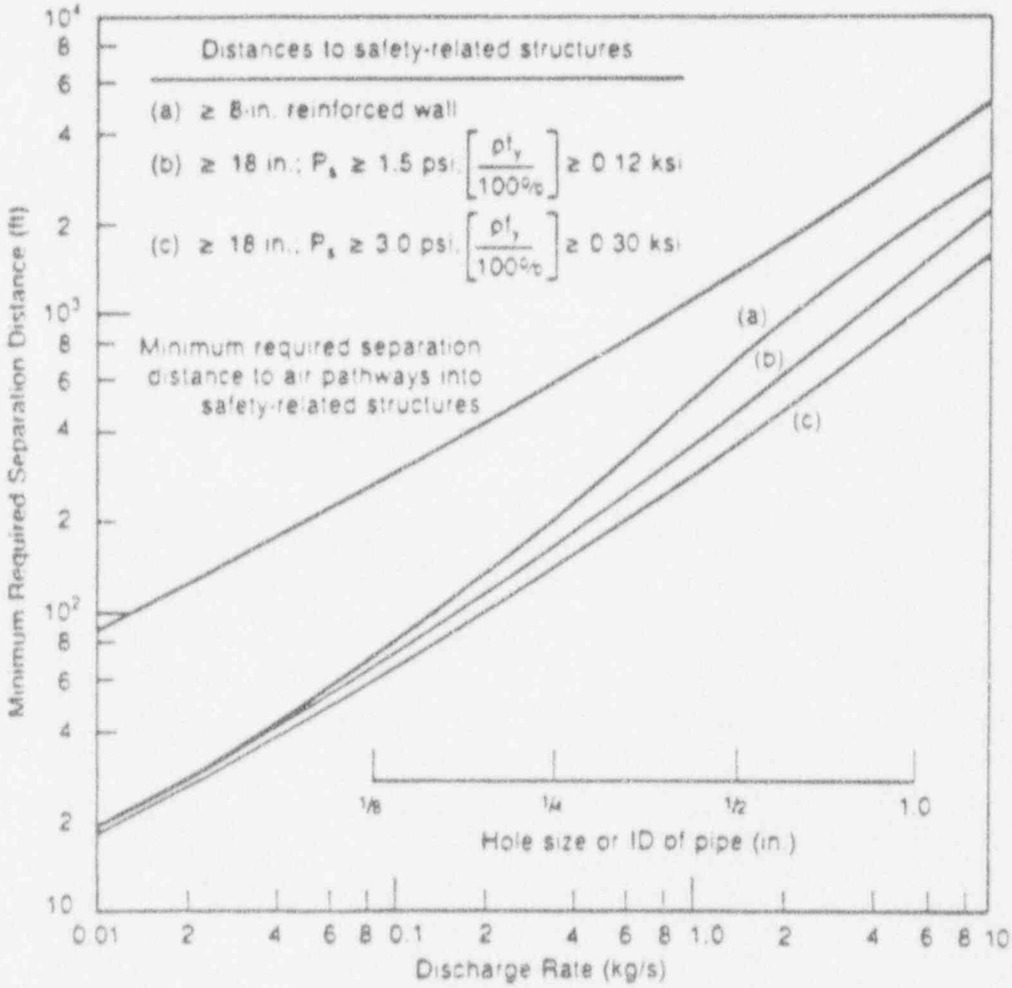

Figure 4-7. Minimum required separation distance versus hole size and discharge rate for liquid releases from 150 psig liquid hydrogen storage tank F weather stability, 1 m/s wind velocity

Section 6

OPERATION, MAINTENANCE, AND TRAINING

This section gives recommendations to the operating utility for operation, maintenance, and training in order to meet the design intent of the hydrogen water chemistry (HWC) system.

The operation of a HWC system will require operator and chemistry personnel attention. Because of the radiation increases that result from employing this system, an awareness of ALARA principles is required by all plant personnel. This system could also have an effect on the off-gas system and the plant fire protection program.

6.1 OPERATING PROCEDURES

Written procedures describing proper valving alignment and sequence for any anticipated operation should be provided for each major component and system process. Check-off lists should be developed and used for complex or infrequent modes of operation. Operating procedures should be considered for the following operations:

1. Hydrogen addition system startup, normal operation, shutdown and alarm response.
2. Material (gas or liquid) handling (filling of storage tanks) operations that are consistent with the supplier's recommendations.
3. Purging of hydrogen and oxygen lines.
4. Operation of on-site gas generation system (if appropriate).
5. Fire protection or safety measures for hydrogen- or oxygen-enhanced fires and hydrogen or oxygen spills.
6. Calibration and maintenance procedures as recommended by equipment or gas suppliers.
7. Routine inspection of HWC system equipment.
8. Adjustment of the main steam line radiation monitor setpoints (if appropriate).

6.1.1 Integration Into Existing Plant Operation Procedures

Where appropriate, operation of the HWC system shall be incorporated into normal plant procedures such as plant startup and shutdown.

6.1.2 Plant-Specific Procedures

Appropriate procedures shall be developed to provide guidance for plant operators when operation of the HWC system necessitates operation of an existing system in a different mode or raises new concerns. Areas which should be considered are:

1. Operation of the off-gas system
2. Possible off-gas fires

6.1.3 Radiation Protection Program

Operation of an HWC system results in an increase in radiation levels wherever nuclear steam is present. The radiation protection program shall be reviewed and appropriate changes made to compensate for these increased radiation levels.

The following guidelines are established to ensure that radiological exposures to both plant personnel and the general public are consistent with ALARA requirements. Compliance with these requirements minimizes radiologically significant hazards associated with HWC implementation. The operation of a hydrogen addition system may cause a slight reduction in the off-gas delay time due to the increase in the flow rate of noncondensables resulting from the excess oxygen added. This may slightly increase plant effluents and should be reviewed on a plant-specific basis.

6.1.3.1 ALARA Commitment. Permanent hydrogen water chemistry systems and programs will be designed, installed, operated, and maintained in accordance with the provisions of Regulatory Guides 8.8 and 8.10 to assure that occupational radiation exposures and doses to the general public will be "as low as reasonably achievable."

6.1.3.2 Initial Radiological Survey. A comprehensive radiological survey should be performed with hydrogen injection to quantify the impact of hydrogen water chemistry on the environs dose rates, both within and outside the plant. This survey should be used to determine if significant radiation changes occur within the plant and at the site boundary. Based upon the magnitude of the change, it operating on HWC, the dissolved oxygen concentration drops below 20 ppb, an evaluation should be made to determine if there is increased corrosion or metals tranport, or if other factors relating to such a reduced oxygen concentration need to be considered. If this evaluation determines that oxygen injection is necessary, a system should be designed using the guidance provided in sections 2.3.2 and 3.4 of this report.

6.1.5 Fuel Surveillance Program

No significant effect of hydrogen injection on fuel performance has been observed, nor is expected. However, since in-reactor experience with hydrogen water chemistry is limited, utilities should consider the fuel surveillance programs recommended by their fuel suppliers.

6.2 MAINTENANCE

A preventative maintenance program should be developed and instituted to ensure proper equipment performance to reduce unscheduled repairs. All maintenance activities should be carefully planned to reduce interference with station operation, assure industrial safety, and minimize maintenance personnel exposure. Written procedures should be developed and followed in the performance of maintenance work. They should be written with the objective of protecting plant personnel from physical harm and radiation exposure, and of reducing hydrogen addition system downtime. Radiation exposure should be reduced by shortening the time required in a high radiation field and by reducing its intensity by turning off the HWC system or other means during the maintenance period.

All excess flow check valves used for hydrogen line break protection shall be periodically tested to assure they will function properly.

6.3 TRAINING

In order for the HWC system to maintain its system integrity and to provide the expected benefits from its use, the system must be operated correctly. The most effective means of reducing the potential of operator error is through proper training.

Training should be provided to:

1. Instruct operators on the function, theory and operating characteristics of the system and all its major system components.

2. Advise operators of the consequences of component malfunctions and misoperation and provide instruction as to appropriate corrective actions to be taken.

3. Advise operations and maintenance personnel of the potential hazards of gases in the system, and provide instruction as to appropriate procedures for their handling.

4. Instruct emergency response personnel on appropriate procedures for handling fires or personnel injuries involving spills or releases of H_2 or O_2 liquid and gases.

5. Instruct plant personnel on the expected radiation changes due to the operation of the HWC system and the appropriate ALARA practices to be taken to minimize dose.

6. Instruct appropriate personnel on the benefits of HWC.

7. Advise maintenance and construction personnel of the routing of hydrogen lines and of the appropriate protective actions to be taken when working near these lines.

Periodic training should be provided to reinforce information described above and to communicate information regarding any modifications, procedural changes, or incidents.

6.4 IDENTIFICATION

In order to aid plant personnel in identifying hydrogen and oxygen lines, these lines should be color coded as required by ANSI A13.1.

6.5 REFERENCES

1. "Environmental Impact of Hydrogen Water Chemistry." EPRI Hydrogen Water Chemistry Workshop, Atlanta, Georgia, December 1984.

2. "Inspection of BWR Stainless Steel Piping." NRC Generic Letter 84-11, April 19, 1984.

3. "Report of the United States Nuclear Regulatory Commission Piping Review Committee." NUREG-1061, Volume 1, August 1984.

4. BWR Hydrogen Water Chemistry Guidelines: 1987 Revision. NP-4947-SR-LD. Palo Alto, Calif.: Electric Power Research Institute, to be published.

Section 7

SURVEILLANCE AND TESTING

7.1 SYSTEM INTEGRITY TESTING

In addition to the testing required by the applicable design codes, completed process systems which will contain hydrogen shall be leak tested with helium or a soap solution as appropriate prior to initial operation of the system. All components and joints shall be so tested in the fabrication shop or after installation, as appropriate. Appropriate helium leak tests shall be performed on portions of the system following any modifications or maintenance activity which could affect the pressure boundary of the system.

7.2 PREOPERATIONAL AND PERIODIC TESTING

Completed systems should be tested to the extent practicable to verify the operability and functional performance of the system. Proper functioning of the following items should be verified:

1. Trip and alarm functions per table 2-2.
2. Gas purity, if generated on site.
3. Safety features.
4. Excess flow check valves.
5. System controls and monitors per table 2-2.

A program should be developed for periodic retesting to verify the operability and the functional performance of the system.

Section 9

QUALITY ASSURANCE

Although the HWC system is non-nuclear safety related, the design, procurement, fabrication and construction activities shall conform to the quality assurance provisions of the codes and standards specified herein. In addition, or where not covered by the referenced codes and standards, the following quality assurance features shall be established.

9.1 SYSTEM DESIGNER AND PROCURER

1. Design and Procurement Document Control--Design and procurement documents shall be independently verified for conformance to the requirements of this document by individual(s) within the design organization who are not the originators of the design and procurement documents. Changes to design and procurement documents shall be verified or controlled to maintain conformance to this document.

2. Control of Purchased Material, Equipment and Services--Measures shall be established to ensure that suppliers of material, equipment and construction services are capable of supplying these items to the quality specified in the procurement documents. This may be done by an evaluation or a survey of the suppliers' products and facilities.

3. Handling, Storage, and Shipping--Instructions shall be provided in procurement documents to control the handling, storage, shipping and preservation of material and equipment to prevent damage, deterioration, and reduction of cleanliness.

9.2 CONTROL OF HYDROGEN STORAGE AND/OR GENERATION EQUIPMENT SUPPLIERS

In addition to the requirements in section 9.1, the system designer should audit the design and manufacturing documents of the equipment supplier to assure conformance to the procurement documents. The system designer shall specify specific factory tests to be performed which will assure operability of the supplier's equipment. The system designer or his representative should be present for the factory tests.

9.3 SYSTEM CONSTRUCTOR

1. Inspection--In addition to code requirements, a program for inspection of activities affecting quality shall be established and executed by, or for, the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. This shall include the visual inspection of components prior to installation for conformance with procurement documents and visual inspection of items and systems following installation, cleaning, and passivation (where applied).

2. Inspection, Test and Operating Status--Measures shall be established to provide for the identification of items which have satisfactorily passed required inspections and tests.

3. Identification and Corrective Action for Items for Nonconformance--Measures shall be established to identify items of nonconformance with regard to the requirements of the procurement documents or applicable codes and standards and to identify the remedial action taken to correct such items.

Enclosure 2

CRGR Item IV.B. Contents of Packages Submitted to CRGR
(Rev. 4, Stello to List 042387, dcs 41860 342 ff)

The following requirements apply for proposals to reduce existing requirements or (regulatory) positions as well as proposals to increase requirements or (regulatory) positions. Each package submitted to the CRGR for review shall include twenty (20) copies of the following information:

SUBJECT: EVALUATION OF HAZARDS ASSOCIATED WITH THE USE OF HYDROGEN, PROPANE, OR NITROGEN AT NUCLEAR POWER PLANTS

Question:

I. The proposed generic requirement as it is proposed to be sent out to licensees.

Response:

The actions being requested of licensees and applicants are spelled out in the proposed bulletin enclosed with this review package.

Question:

II. Draft staff papers or other underlying staff documents supporting the requirements or staff (regulatory) positions. (A copy of all materials referenced in the document shall be made available upon request to the CRGR staff. Any committee member may request CRGR staff to obtain a copy of any referenced material for his or her use.)

Response:

An acceptance criteria for the safe usage of hydrogen onsite is found in "Guidelines For Permanent BWR Hydrogen Water Chemistry Installation," 1987 Revision, dated September 1987. An NRC letter from James E. Richardson to G. H. Neils, Chairman of Regulatory Advisory Committee, BWR Owners Group II for Intergranular Stress Corrosion Cracking Licensing Research, dated July 13, 1987, accepts for referencing the Guidelines Licensing Topical Report. A copy of this letter and the applicable Safety Evaluation Report have been enclosed for information. Information Notice Nos. 87-20 and 89-44 are also enclosed. The EPRI Guidelines, a licensing topical report, contains acceptable methods of complying with GDC 3 and GDC 19 in regard to hydrogen storage and usage onsite.

Branch Technical Position (BTP) CMEB 9.5-1 addresses hydrogen storage and usage in safety-related areas in a limited manner.

Question:

III. Each proposed requirement or staff (regulatory) position shall contain the sponsoring office's position as to whether the proposal would increase requirements or staff (regulatory) positions, implement existing requirements or staff (regulatory) positions, or would relax or reduce existing requirements or staff (regulatory) positions.

Response:

The actions requested by this bulletin are intended to assure that onsite hydrogen and propane facilities have (1) a separation distance sufficient to prevent damage to safety-related structures in the event of an explosion, (2) a separation distance sufficient to prevent a flammable or explosive gas mixture from entering the control room air intakes in the event of a pipe break, (3) a separation distance sufficient to prevent large quantities of nitrogen from entering the control room air intakes and incapacitating the operators, and (4) been designed to prevent damage to safety-related equipment in the auxiliary and turbine buildings. The above requirements are covered in GDC 3 and GDC 19 of 10 CFR Part 50, Appendix A.

Question:

IV. The proposed method of implementation along with the concurrence (and any comments) of OGC on the method proposed.

Response:

A. The proposed method of implementation is via a bulletin that requires addressees, within 60 days of receipt of the bulletin, to advise the NRC of what actions they propose to take.

B. The Office of the General Counsel (OGC) has reviewed the proposed bulletin and its comments have been incorporated.

Question:

V. Regulatory analyses generally conforming to the directives and guidance of NUREG/CR-0058 and NUREG/CR-3568. (Make analyses sufficient to address the Paperwork Reduction Act, the Regulatory Flexibility Act and Executive Order 12201.)

Response:

A regulatory analysis with a value-impact analysis is not required because the actions requested in this bulletin would bring licensees into regulatory compliance with GDC 3 and 19 and do not impose new requirements. This request for information was approved by the Office of Management and Budget under Blanket Clearance Number 31500011 as meeting requirements of the Paperwork Reduction Act and Executive Order 12201. Sufficient hours are included in the NRC budget for this request.

As this is not a rulemaking action, the Regulatory Flexibility Act does not apply.

Question:

VI. Identification of the category of reactor plants to which the generic requirement or staff position is to apply (that is, whether it is to apply to new plants only, new operating licenses (OLs) only, OLs after a certain date, OLs before a certain date, all OLs, all plants under construction, all plants, all water reactors, all PWRs only, some vendor types, some vintage types such as BWRs 6 and 4, jet pump and non-jet pump plants).

Response:

This proposed bulletin applies to all nuclear power plants (both operating and under construction).

Question:

VII. For each such category of reactor plants, an evaluation which demonstrates how the action should be prioritized and scheduled in light of other ongoing regulatory activities. The evaluation shall document for consideration information available concerning any of the following factors as may be appropriate and any other information relevant and material to the proposed action.

Response:

Response to this item is not required pursuant to Revision 4 of the CRGR Charter, Section III.D, as the recommended actions of the proposed bulletin are intended to bring facilities into conformance with the rules and regulations of the Commission.

Question:

VIII. For each evaluation conducted pursuant to 10 CFR 50.109, the proposing office director's determination, together with the rationale for the determination based on the considerations of all the above, that:

A. There is a substantial increase in the overall protection of public health and safety or the common defense and security to be derived from the proposal.

Response:

Hydrogen and propane storage facilities with inadequate separation distance from safety-related structures and air intakes can adversely affect the performance of safety-related equipment and safe operation of the plant in the event of a hydrogen or propane explosion or a flammable or explosive gas mixture entering the control room air intakes from a pipe break or storage facility leakage. Hydrogen lines in auxiliary and turbine buildings can affect the performance of safety-related equipment in the area of a potential undetected leak and subsequent fire or explosion. Moderate leakage from nitrogen storage facilities with inadequate separation distance from control room air intakes can incapacitate the operators. Actions proposed by the staff in this bulletin would prevent the above potential safety problems. Thus, the staff concludes that the actions requested in the bulletin will result in a substantial increase in the overall protection of the public health and safety.

Question:

B. The direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection.

Response:

Although it may not be representative of the impact of the bulletin on each addressee, the estimated total cost for relocating the hydrogen supply system at the Trojan Nuclear Plant is approximately $400,000. This includes installation of a new bulk hydrogen storage facility south of the cooling tower and distribution piping from the new facility to the Turbine Building roof where it will tie into the existing supply to the volume control tank and generator cooling system. The cost information is provided in response to this CRGR package item. Since Section 109(a)(4) of 10 CFR Part 50 and Section III D of the CRGR charter state that costs are not required when generic modifications are necessary to bring facilities into compliance, costs should not be considered in reviewing the bulletin. In addition, the reports that the bulletin requires addressees submit do not require NRC staff review and approval; therefore, NRC staff effort is minimized.

Question:

IX. For each evaluation conducted for proposed relaxations or decreases in current requirements or staff (regulatory) positions, the proposing office director's determination, together with the rationale for the determination based on the considerations of all the above, that the public health and safety and the common defense and security would be adequately protected if the proposed reduction in requirements or (regulatory) positions were implemented, and the cost savings attributed to the action would be substantial enough to justify taking the action.

Response

This item is not applicable to the proposed bulletin as no relaxation or decrease in current requirements is being proposed.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

April 27, 1989

NRC INFORMATION NOTICE NO. 89-44: HYDROGEN STORAGE ON THE ROOF OF THE CONTROL ROOM

Addressees:

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose:

This information notice is provided to alert recipients to potential generic problems pertaining to the storage of hydrogen in the vicinity of safety-related structures and air pathways into safety-related structures. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

During the Region V Chemistry Team Inspection at the Trojan Nuclear Plant the week of April 17, 1989, the inspectors identified a potential safety problem concerning the location of the hydrogen storage facility. Hydrogen is used on pressurized water reactor (PWR) plants for (1) providing a cover gas in the volume control tank, and (2) for cooling the main turbine generator. At boiling water reactor (BWR) plants, hydrogen is also used for cooling the main turbine generator and for injection into the feed system for plants which have implemented hydrogen water chemistry. The Trojan hydrogen storage facility is located on the control room roof which is 30-inch-thick reinforced concrete. The following potential safety problems were identified during the Region V Chemistry Team Inspection:

1. Leakage of hydrogen gas from the storage facility in proximity to the air intakes to the control room ventilation and emergency pressurization system may introduce a flammable or explosive gas mixture into the control room. Because the hydrogen storage facility, containing four 8,000-scf hydrogen tanks at up to 2450 psig, is Seismic Category II, a seismic event may result in a hydrogen leak. Furthermore, the pressure relief valves in the hydrogen facility exhaust downward to within 6 inches of the control room roof in the vicinity of the control room ventilation system air intakes. It was also noted that six 8,000-scf nitrogen tanks were located

in the vicinity of the control room air intakes. Nitrogen leakage and dispersion into the air intakes may lead to incapacitation of the control room operators.

2. A detonation of a hydrogen storage tank (energy equivalent to 217 pounds of TNT) may structurally damage and affect performance of safety-related equipment on the control room roof such as the ventilation system intake and exhaust structure, the emergency pressurization system, and equipment in the control room itself.

3. An explosion of the hydrogen delivery truck that provides hydrogen to the facility through a fill line located at ground level on the wall of the auxiliary building may structurally damage safety-related component cooling water pumps and radwaste storage tanks located inside the auxiliary building and in the vicinity of the hydrogen fill line.

Discussion:

The topical report "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," 1987 Revision, EPRI NP-5283-SR-A was reviewed and accepted by NRC. NRC's approval letter, dated July 13, 1987, states that this topical report may be useful in providing industry guidance for the design, operation, maintenance, surveillance, and testing of hydrogen systems for (1) providing a cover gas in the PWR volume control tank and (2) for cooling the main turbine generator. In addition, NRC Information Notice No. 87-20, "Hydrogen Leak In Auxiliary Building," dated April 20, 1987, indicated that the NRC was then reviewing the EPRI/BWROG topical report (EPRI NP-5283-SR-A). The Trojan plant hydrogen facility does not meet these guidelines from the standpoint of (1) the separation distance needed between a hydrogen pipe break and the control room ventilation intake to prevent buildup of a flammable or explosive gas mixture inside the control room, and (2) the separation distance needed to prevent damage to safety-related structures resulting from the explosion of an 8,000-scf hydrogen tank.

Related Generic Communications:

NRC Information Notice No. 87-20, "Hydrogen Leak In Auxiliary Building," dated April 20, 1987, discusses leakage of hydrogen from a volume control tank globe valve in the auxiliary building.

NUREG/CR-3551, ORNL/NOAC-214 "Safety Implications Associated With In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants," dated May 1985, provides information useful in considering hazards and methods to ensure the safe handling of pressurized gases, including hydrogen.

EPRI NP-5283-SR-A, "Guidelines For Permanent BWR Hydrogen Water Chemistry Installations" - 1987 Revision, dated September 1937, is a topical report approved by the NRC that provides industry guidance for the design, operation, maintenance, surveillance, and testing of hydrogen systems. It was also recommended by the NRC for use on hydrogen systems for (1) providing a cover gas in the PWR volume control tank, and (2) for cooling the main turbine generator.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the technical contact listed below or the Regional Administrator of the appropriate regional office.

Charles E. Rossi

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contact: Frank J. Witt, NRR
(301) 492-0823

Attachment: List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
88-82, Supp. 1	Torus Shells with Corrosion and Degraded Coatings in BWR Containments	5/2/89	All holders of OLs or CPs for BWRs.
89-43	Permanent Deformation of Torque Switch Helical Springs in Limitorque SMA-Type Motor Operators	5/1/89	All holders of OLs or CPs for nuclear power reactors.
88-97, Supp. 1	Potentially Substandard Valve Replacement Parts	4/28/89	All holders of OLs or CPs for nuclear power reactors.
89-42	Failure of Rosemount Models 1153 and 1154 Transmitters	4/21/89	All holders of OLs or CPs for nuclear power reactors.
89-41	Operator Response to Pressurization of Low-Pressure Interfacing Systems	4/20/89	All holders of OLs or CPs for nuclear power reactors.
88-75, Supplement 1	Disabling of Diesel Generator Output Circuit Breakers by Anti-Pump Circuitry	4/17/89	All holders of OLs or CPs for nuclear power reactors.
89-40	Unsatisfactory Operator Test Results and Their Effect on the Requalification Program	4/14/89	All holders of OLs or CPs for nuclear power reactors.
89-39	List of Parties Excluded from Federal Procurement or Non-Procurement Programs	4/5/89	All holders of OLs or CPs for nuclear power reactors.
89-38	Atmospheric Dump Valve Failures at Palo Verde Units 1, 2, and 3	4/5/89	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

SSINS No.: 6835
IN 87-20

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

April 20, 1987

NRC INFORMATION NOTICE NO. 87-20: HYDROGEN LEAK IN AUXILIARY BUILDING

Addressees:

All nuclear power reactor facilities holding an operating license or a construction permit.

Purpose:

This notice is to alert addressees of the potential for a hydrogen leak in portions of the plant where the potential for the leak may not have been adequately considered. Recipients are expected to review the information for applicability to their facilities and consider actions, if appropriate, to preclude similar problems occurring at their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On February 20, 1987, the Vogtle nuclear power plant reported a hydrogen leak inside the auxiliary building. This plant was recently licensed, had never been critical, and was in cold shutdown at the time of the event.

The discovery of this problem was as a result of an unassociated event involving the activation of a chlorine monitor in the control building. When additional samples indicated no chlorine gas, the shift supervisor ordered further investigation into other plant areas. Because there was no installed detection equipment, portable survey instruments were used to determine gaseous mixtures. Hydrogen was detected in the auxiliary building and indicated about 20 to 30 percent of the lower flammability limit (LFL) for hydrogen. A level of about 30 percent of LFL corresponds to about 1.2 percent hydrogen by volume. This reading was erroneously reported to the control room as 20 to 30 percent hydrogen by volume. The on-shift supervisor declared an unusual event (UE) with a subsequent report to the NRC via the emergency notification system (ENS).

When hydrogen was discovered in the auxiliary building, the licensee isolated the cryogenic hydrogen skid outside the turbine building and soon located the source of the leak as packing on a globe valve in a small line to the volume control tank (VCT). The licensee opened doors that quickly caused the hydrogen to dissipate. The globe valve was of a conventional design and had no special packing. The globe valve was located in a vertical pipe chase where little

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ventilation was present because of ongoing HVAC testing. Besides being used as a cover gas in the VCT, hydrogen from the skid also is used in the plants waste gas system and to cool the generator.

Discussion:

The lessons of this event fall into five categories: (1) proper in-plant communications during events, (2) proper valve application for use with hydrogen, (3) excess flow check valve set point, (4) heating ventilation and air conditioning (HVAC) maintenance and flow testing, and (5) hydrogen line routing.

The licensee is examining ways to improve communications in the plant during events and the training of personnel in reading portable instruments.

As another corrective measure, the licensee is examining the use of other types of valves, such as valves with a diaphragm or bellows rather than conventional stem packing, in lines containing hydrogen.

The licensee also is examining the set point for the excess flow check valves in the hydrogen lines. These check valves are designed to limit the flow of hydrogen in the event of a large leak so that when combined with proper ventilation in rooms with hydrogen lines, hydrogen levels would remain within specified limits throughout the plant.

This plant had HVAC flow balancing problems during the preparation for plant startup. Generally HVAC flow balance is based on the heat loads and the resultant room temperatures under normal and accident conditions. However, this event demonstrates that hydrogen concentrations also may need to be considered to set a lower limit on the ventilation in rooms that contain hydrogen lines.

Although this licensee has reexamined the routing of hydrogen lines throughout the auxiliary building and found no problems, licensees with older plants may not have examined this question in detail.

The NRC staff is currently reviewing an EPRI/BWROG topical report titled "Guidelines for Permanent BWR Hydrogen Water Chemistry Installation," 1987 revision. Included in this document are guidelines for design, operation, maintenance, surveillance, and testing of hydrogen supply systems.

Other Recent Reactor Events Involving Hydrogen

On March 3, 1987 an unusual event was reported at Waterford Unit 3 plant. While unloading hydrogen from a truck into the storage tank, the storage tank rupture disc failed and a deflagration and fire ensued. The fire burned itself out in about an hour with no apparent damage to the storage facility.

On January 12, 1987, an explosive mixture of hydrogen and oxygen was discovered in the number 1 holdup tank of the gaseous radwaste system at Zion Unit 1. Prompt action was taken to isolate the tank and dilute the gaseous content with a nitrogen purge to reduce the hydrogen concentration below explosive limits. Investigation showed that the holdup tank was placed in service on January 6, 1987. However, the tank was left isolated from the automatic waste gas analyzer until January 12, 1987. This violated the technical specifications requiring daily analysis of the waste gas system for oxygen and hydrogen.

A report that may be useful in considering hazards and some methods for improving the safe handling of pressurized gas is NUREG/CR-3551, ORNL/NOAC-214 "Safety Implications Associated with In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants," published in May 1985.

No specific action or written response is required by this information notice. If you have questions about this matter, please contact the Regional Administrator of the appropriate NRC regional office or this office.

Charles E. Rossi

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Eric Weiss, AEOD
(301) 492-9005

Frank Witt, NRR
(301) 492-9440

Attachment: List of Recently Issued IE Information Notices

Heltemes

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

JUL 2 1 1989

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee To Review Generic Requirements

FROM: James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: DRAFT BULLETIN 88-10, SUPPLEMENT 1:
NONCONFORMING MOLDED-CASE CIRCUIT BREAKERS

The Office of Nuclear Reactor Regulation (NRR) transmitted the subject bulletin supplement to you by memorandum dated July 5, 1989, and requested that review by the Committee to Review Generic Requirements (CRGR) be waived.

Your staff has indicated that CRGR would like to formally review the bulletin supplement. Accordingly, we are transmitting the information (see enclosures) required for CRGR review.

NRR requests that review of this package be scheduled at CRGR's earliest convenience. Should you have any questions, please contact my office.

Frank J. Miraglia

for James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Memorandum from J. H. Sniezek to E. L. Jordan dated July 5, 1989, Transmitting Draft NRC Bulletin No. 88-10, Supplement 1
2. CRGR Package (Item IV.B.)
3. References Submitted to CRGR
4. CRGR Package for Bulletin No. 88-10
5. NRC Bulletin No. 88-10

CONTACT: Jaime Guillen, NRR
492-1170

8907280350 3PP

JUL 2 1 1989

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee To Review Generic Requirements

FROM: James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

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Original signed by
Frank J. Miraglia

James H. Sniezek, Deputy Director
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5. NRC Bulletin No. 88-10

CONTACT: Jaime Guillen, NRR
492-1170

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JGuillen R/F			

*SEE PREVIOUS CONCURRENCES

		DD/NRR	ADT/NRR	D/DOEA:NRR
		JHSniezek	FJMiraglia	CERossi
		07/21/89	07/ 21/89	07/21/89
*OGCB:DOEA:NRR	*RVIB:DRIS:NRR	*C/RVIB:DRIS:NRR	*D/DRIS:NRR	C/OGCB:DOEA:NRR
JGuillen	UPotapovs	EWBrach	BKGrimes	CHBerlinger
07/19/89	07/19/89	07/ /89	07/ /89	07/21/89

JUL 0 5 1989

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

FROM: James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: DRAFT NRC BULLETIN NO. 88-10, SUPPLEMENT 1: NONCONFORMING MOLDED-CASE CIRCUIT BREAKERS

The Office of Nuclear Reactor Regulation (NRR) has conducted a preliminary review of responses required by NRC Bulletin No. 88-10 and has identified several common reporting deficiencies. In addition, the NRC staff has received requests for positions on specific issues that were not explicitly addressed in Bulletin No. 88-10. As a result, NRR has prepared the enclosed supplement to Bulletin No. 88-10.

The supplement requests that addressees 1) review their written reports submitted to the NRC, 2) verify that the responses meet the bulletin provisions and are consistent with positions delineated in the supplement, and 3) prepare and retain documentation for possible audit that indicates that they have performed the requested review. In addition, the supplement requires that addressees submit any appropriate corrections to their previous responses to Bulletin No. 88-10.

The supplement does not contain any new requests or requirements that were not part of the staff's original intent and simply elaborates on some reporting deficiencies identified in the written reports submitted to the NRC; therefore, NRR proposes that the review by the Committee to Review Generic Requirements (CRGR) be waived. Please inform us of your decision as soon as possible so that we can issue the bulletin supplement promptly or prepare an appropriate CRGR package. This supplement has been reviewed by OGC and their comments have been addressed.

If you have any questions regarding this matter, please contact my office.

Original signed by
James H. Sniezek

James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

CONTACT: Jaime Guillen, NRR
492-1170

~~8907140151~~ 10 pp

DISTRIBUTION w/enclosure

TEMurley, NRR	JHSniezek, NRR	FJMiraglia, NRR
FGillespie, NRR	CERossi, NRR	CHBerlinger, NRR
JGuillen, NRR	JACalvo, NRR	WDLanning, NRR
GHolahan, NRR	SAVarga, NRR	DCrutchfield, NRR
LCShao, NRR	FCongel, NRR	JWRoe, NRR
BKGrimes, NRR	JGPartlow, NRR	AThadani, NRR
JRichardson, NRR	SATreby, OGC	WGKennedy, EDO
JHConran, AEOD	WTRussell, RI	SEbneter, RII
ABDavis, RIII	RDMartin, RIV	JBMartin, RV
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*OGCB:DOEA:NRR	*C/OGCB:DOEA:NRR	*D/DOEA:NRR	*ADT/NRR	*DD/NRR
JGuillen	CHBerlinger	CERossi	FMiraglia	JHSniezek
06/22/89	06/26/89	06/27/89	06/29/89	07/ 5/89

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

DRAFT

July xx, 1989

NRC BULLETIN NO. 88-10, SUPPLEMENT 1: NONCONFORMING MOLDED-CASE CIRCUIT BREAKERS

Addressees:

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose:

The purpose of this bulletin is to inform addressees that based on a preliminary review of responses to NRC Bulletin No. 88-10, the NRC staff has determined that many responses do not adequately satisfy the provisions of Bulletin No. 88-10 and that some addressees may need to take additional actions. This supplement also provides specific examples of common deficiencies identified during the preliminary review of responses.

Description of Circumstances:

NRC Bulletin No. 88-10 was issued on November 22, 1988, to request that addressees take actions to provide reasonable assurance that molded-case circuit breakers (CBs) purchased for use in safety-related applications perform their safety functions. In addition, the bulletin required that addressees submit certain information to the NRC regarding CBs that could not be traced to the circuit breaker manufacturer (CBM).

An NRC staff review of the written reports submitted by addressees in accordance with Bulletin No. 88-10 revealed several common deficiencies. In addition, the NRC staff has received requests for positions on specific issues that were not explicitly addressed in Bulletin No. 88-10. The NRC analyses and positions on these issues are provided in this supplement.

During the preparation of this supplement, the NRC received comments from the National Electrical Manufacturers Association (NEMA) and the Nuclear Management and Resources Council (NUMARC). NEMA reaffirmed its position that neither the tests delineated in Bulletin No. 88-10, a visual inspection, nor a combination of testing and inspection, are adequate to ensure the performance of nontraceable CBs. Similarly, NUMARC raised concerns about and advised against the use of nontraceable CBs from known refurbishers, regardless of whether or not they have passed the tests delineated in Bulletin No. 88-10. However, the NRC judgement on the adequacy of bulletin testing to justify continued use of nontraceable CBs remains as stated in Bulletin 88-10.

NRC Positions:

1. If CBs are traceable to an original plant construction order and the CBs were received prior to August 1983, there is reasonable assurance that the CBs are acceptable and no additional traceability is required.

2. Visual inspection and physical examination of the CBs by the CBM is not considered adequate to meet the traceability requirements of Bulletin No. 88-10. Although visual inspection and physical examination by the CBM may provide a reasonable basis that the CBs have not been opened or altered in a substantial way, there is no reasonable assurance that the CBs have not been previously used or subjected to service conditions that may have adversely affected the performance capabilities of the CBs.

3. Item 4 of the actions requested in Bulletin No. 88-10 applies only to CBs that were purchased and installed after August 1, 1983.

4. If an addressee identifies any CBs as nontraceable during the review requested by Bulletin No. 88-10, it should take appropriate corrective actions as required by Criterion XVI of 10 CFR Part 50, Appendix B. As part of these corrective actions, the NRC expects addressees to assess the acceptability of all installed CBs that were procured under the same purchase orders as the nontraceable CBs, regardless of whether or not they are subject to Bulletin No. 88-10.

5. Nontraceable CBs that were installed or being maintained as stored spares as of August 1, 1988, and that successfully pass all tests specified in Attachment 1 of Bulletin No. 88-10 are considered acceptable for use only as replacements for safety-related CBs that are found to be nontraceable during the review requested by Bulletin No. 88-10. These breakers may not be used as safety-related replacements during other activities such as planned plant modifications or routine maintenance.

6. For CBs stored as spares that were not procured directly from the CBM, each individual CB should be reviewed in order to establish proper traceability, regardless of the number of CBs.

7. All safety-related CBs from the same procurement order are considered traceable if the order was procured directly from a CBM having a quality assurance program in accordance with 10 CFR Part 50, Appendix B, if the CBM has been audited by the addressee in accordance with Appendix B, if the CBs were ordered as safety-related, and if documented evidence has been furnished to the addressee, such as a certificate of compliance. However, if safety-related CBs were procured from a vendor other than the CBM, a certificate of compliance by itself is not considered an adequate

basis for establishing traceability. In such cases, traceability of individual procurement orders should be established through the review of procurement or shipping records back to the CBM. Telephone discussions with the CBM or vendor are not acceptable for establishing a basis for traceability. Traceability to a warehouse facility controlled by the CBM is considered equivalent to traceability to the CBM.

Actions Requested:

In response to the aforementioned circumstances, addressees are requested to perform the following actions within 90 days from the receipt of this bulletin:

1. Review written reports submitted to the NRC in accordance with Bulletin No. 88-10 and verify that the responses meet the bulletin provisions and are consistent with the above NRC positions.

2. Prepare and retain documentation for possible audit that indicates that item 1 of the actions requested has been performed as requested.

Reporting Requirements:

Addressees are required to provide any appropriate corrections to previous responses to Bulletin No. 88-10.

The NRC may conduct inspections at selected nuclear power plant sites in order to verify that issues associated with Bulletin No. 88-10 and this supplement have been adequately resolved.

The written reports required above shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended. In addition, a copy shall be submitted to the appropriate Regional Administrator.

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires December 31, 1989. The estimated burden hours, which includes the original bulletin requests, is 1,000 to 10,000 person-hours per plant response, including assessment of these requirements, searching data sources, testing, analyzing the data, and preparing the required reports. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden to the Records and Reports Management Branch, Division of Information Support Services, Office of Information Resources Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Paperwork Reduction Project (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

If you have any questions regarding this matter, please contact one of the technical contacts listed below or the Regional Administrator of the appropriate NRC regional office.

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Uldis Potapovs, NRR
(301) 492-0984

Jaime Guillen, NRR
(301) 492-1170

Attachment: List of Recently Issued NRC Bulletins

If you have any questions regarding this matter, please contact one of the technical contacts listed below or the Regional Administrator of the appropriate NRC regional office.

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Uldis Potapovs, NRR
(301) 492-0984

Jaime Guillen, NRR
(301) 492-1170

Attachment: List of Recently Issued NRC Bulletins

*SEE PREVIOUS CONCURRENCES
*D/DOEA:NRR
CERossi
06/26/89

*D/DRIS:NRR	*C/OGCB:DOEA:NRR	*RPB:ARM	*D/DEST:NRR	*AD/DEST:NRR
BKGrimes	CHBerlinger	TechEd	LCShao	AThadani
06/23/89	06/26/89	06/01/89	06/22/89	06/21/89
*OGCB:DOEA:NRR	*RVIB:DRIS:NRR	*C/RVIB:DRIS:NRR	*C/SELB:DEST:NRR	*OGC
JGuillen	UPotapovs	WBrach	FRosa	SLewis
06/01/89	06/15/89	06/15/89	06/20/89	06/14/89

Enclosure 2

CRGR Item IV.B. Contents of Packages Submitted to CRGR
(Rev. 4, Stello to List 042387, dcs 41860 342 ff)

The following requirements apply for proposals to reduce existing requirements or (regulatory) positions as well as proposals to increase requirements or (regulatory) positions. Each package submitted to the CRGR for review shall include twenty (20) copies of the following information:

SUBJECT: BULLETIN NO. 88-10, SUPPLEMENT 1:
NONCONFORMING MOLDED-CASE CIRCUIT BREAKERS

Question:

I. The proposed generic requirement as it is proposed to be sent out to licensees.

Response:

The proposed generic requirements are delineated in the draft bulletin supplement enclosed with this review package.

Question:

II. Draft staff papers or other underlying staff documents supporting the requirements or staff (regulatory) positions. (A copy of all materials referenced in the document shall be made available upon request to the CRGR staff. Any committee member may request CRGR staff to obtain a copy of any referenced material for his or her use.)

Response:

A. NRC Bulletin No. 88-10: Nonconforming Molded-Case Circuit Breakers

B. Letter from Nuclear Utilities Management and Resources Council (NUMARC) dated May 19, 1989.

C. Letter from National Electric Manufacturers Association (NEMA) dated May 16, 1989.

D. CRGR Package for Bulletin No. 88-10 dated November 14, 1988.

E. NRC Information Notice No. 88-46: Licensee Report of Defective Refurbished Circuit Breakers, and Supplements 1, 2, and 3 thereto.

Question:

III. Each proposed requirement or staff (regulatory) position shall contain the sponsoring office's position as to whether the proposal would increase requirements or staff (regulatory) positions, implement existing requirements or staff (regulatory) positions, or would relax or reduce existing requirements or staff (regulatory) positions.

Response:

The actions requested by this bulletin supplement are intended to assure that addressees meet the provisions and intent of NRC Bulletin No. 88-10. The supplement does not contain any new requests or requirements that were not part of the staff's original intent and simply elaborates on some reporting deficiencies identified in the written reports submitted to the NRC. Staff regulatory positions are not altered by this proposed supplement.

Question:

IV. The proposed method of implementation along with the concurrence (and any comments) of OGC on the method proposed.

Response:

A. The proposed method of implementation is via a bulletin supplement that requests addressees to review their previous responses to Bulletin No. 88-10 and to verify that their responses meet the original bulletin provisions.

B. The Office of the General Counsel (OGC) has reviewed the proposed bulletin supplement and its comments have been incorporated.

Question:

V. Regulatory analyses generally conforming to the directives and guidance of NUREG/CR-0058 and NUREG/CR-3568. (Make analyses sufficient to address the Paperwork Reduction Act, the Regulatory Flexibility Act and Executive Order 12201.)

Response:

A. A regulatory analysis with a value-impact analysis is not required as the bulletin supplement simply requests that addressees meet the original provisions and intent of Bulletin No. 88-10. This request for information was approved by the Office of Management and Budget under Blanket Clearance Number 3150-0011, which expires December 31, 1989, as meeting the requirements of the Paperwork Reduction Act and Executive Order 12201. Sufficient hours are included in the NRC budget for this request. However, it is expected that NRC staff will spend a minimal amount of resources as a result of issuing this bulletin supplement.

B. The Regulatory Flexibility Act does not apply since this is not a rulemaking action.

Question:

VI. Identification of the category of reactor plants to which the generic requirement or staff position is to apply (that is, whether it is to apply to new plants only, new operating licenses (OLs) only, OLs after a certain date, OLs before a certain date, all OLs, all plants under construction, all plants, all water reactors, all PWRs only, some vendor types, some vintage types such as BWRs 6 and 4, jet pump and non-jet pump plants).

Response:

This proposed bulletin supplement applies to all holders of operating licenses and construction permits for nuclear power reactors. The category of reactor plants is consistent with that of the original bulletin.

Question:

VII. For each such category of reactor plants, an evaluation which demonstrates how the action should be prioritized and scheduled in light of other ongoing regulatory activities. The evaluation shall document for consideration information available concerning any of the following factors as may be appropriate and any other information relevant and material to the proposed action.

Response:

Response to this item is not required pursuant to Revision 4 of the CRGR Charter, Section III.D, as the recommended actions of the proposed bulletin are intended to bring facilities into conformance with the rules and regulations of the Commission.

Question:

VIII. For each evaluation conducted pursuant to 10 CFR 50.109, the proposing office director's determination, together with the rationale for the determination based on the considerations of all the above, that:

A. There is a substantial increase in the overall protection of public health and safety or the common defense and security to be derived from the proposal.

Response:

The specific actions in Bulletin No. 88-10 address nonconformances and concerns identified during NRC inspections and provide a reasonable assurance that molded-case CBs will perform their intended functions. The original bulletin also requested addressees to perform certain actions and provide specific information regarding molded-case CBs to assess compliance with existing regulatory requirements. The NRC staff performed a preliminary review of responses submitted by addressees in response to Bulletin No. 88-10 and has determined that many responses do not adequately satisfy the provisions of Bulletin No. 88-10 and that some addressees may need to take additional actions. The enclosed draft bulletin supplement discusses reporting deficiencies identified in the written reports and requests that addressees review their actions to assure that they have met the provisions of Bulletin No. 88-10. This supplement does not contain any new requests or requirements that were not part of the staff's original intent.

Question:

B. The direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection.

Response:

The implementation costs for Bulletin No. 88-10 were estimated to vary between 1,000 to 10,000 person-hours for each addressee depending on the total number of molded-case CBs in stores for or installed in safety-related applications, the number of non-traceable CBs identified, the failure rate of the CBs tested, and replacement costs. The costs for implementing the bulletin supplement are bound by the estimated costs of the original bulletin since addressees that properly carried out the original bulletin provisions will not have to take any additional actions other than to review the bulletin supplement and document that they have reviewed the adequacy of their written reports. Addressees whose actions and reports do not satisfy the original bulletin provisions will have to take additional actions in order to assure that the bulletin provisions are implemented properly and their reports are consistent with NRC positions.

Question:

IX. For each evaluation conducted for proposed relaxations or decreases in current requirements or staff (regulatory) positions, the proposing office director's determination, together with the rationale for the determination based on the considerations of all the above, that the public health and safety and the common defense and security would be adequately protected if the proposed reduction in requirements or (regulatory) positions were implemented, and the cost savings attributed to the action would be substantial enough to justify taking the action.

Response

This item is not applicable to the proposed bulletin as no relaxation or decrease in current requirements is being proposed.

NUMARC

NUCLEAR MANAGEMENT AND RESOURCES COUNCIL

1776 Eye Street, N.W. • Suite 300 • Washington, DC 20006-2496
(202) 872-1280

May 19, 1989

Mr. Frank J. Miraglia
Associate Director for Inspection and
Technical Assessment
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Miraglia:

On Thursday May 11, 1989, NUMARC met with members of the NRC Staff to discuss overall industry responses to Bulletin 88-10, "Nonconforming Molded Case Circuit Breakers." We appreciate such opportunities for continued dialogue with the Staff toward resolving concerns about fraudulent molded case circuit breakers (MCCBs).

The Staff provided for discussion draft Supplement 1 to Bulletin 88-10. With roughly half of utilities in the process of preparing follow-up responses to Bulletin 88-10, we agree that a Bulletin supplement at this time would be an effective method for the Staff to re-emphasize the original intent of the Bulletin and address major inconsistencies in utility responses where it appears the original intent may not be understood. We do, however, have significant reservations about the content of the draft supplement. Our major concerns are summarized below and specific comments on individual supplement items are provided in Attachment 1.

We find that the supplement presents additional requirements not suggested by Bulletin 88-10 that represent significant new burden to many utilities. NUMARC acknowledges that Bulletin 88-10 results could, in the final analysis, suggest that additional efforts may be necessary for industry to definitively resolve concern over nonconforming MCCBs. However, we believe that it is inappropriate to present new requirements at this time since,

1) results of Bulletin 88-10 are not yet complete, and

2) results to date do not appear to provide a basis for further action.

Rather, Supplement 1 should be directed at ensuring existing Bulletin requirements are fully understood by all utilities.

We are concerned that the supplement attempts to alter the Staff's original focus for Bulletin 88-10 and expand its scope without clear justification. Establishing traceability to the CBM was the course chosen by the Staff for providing assurance that replacement MCCBs purchased for safety-related service during the identified period of primary concern (August, 1983 to August, 1988) were not refurbished or previously used.

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Miraglia, Frank J.
May 19, 1989
Page 2

The proposed supplement alters this focus in two critical areas. First, the Bulletin acknowledged that MCCBs purchased during original plant construction were purchased from the circuit breaker manufacturer (CBM) and concluded that such "original" MCCBs were not a concern of this Bulletin. The supplement, in a clear expansion of Bulletin requirements, requires utilities to establish traceability to the CBM for MCCBs purchased as original plant equipment or be required to replace or successfully test them. As noted in Bulletin 88-10, original plant equipment for safety-related service was manufactured and purchased under programs conforming to 10CFR50 Appendix B which provided reasonable assurance that purchased material and equipment conform to the procurement documents. Traceability to the extent it is required by Bulletin 88-10 goes beyond "adequate confidence" as intended by Appendix B, and we feel it is inappropriate to require traceability of original plant MCCBs to their CBM.

Secondly, the Bulletin established a five year review period for installed MCCBs based on anticipated availability of procurement records. The supplement, however, indicates Staff expectation that installed non-traceable MCCBs be subject to Bulletin requirements regardless of purchase date. We must re-emphasize that lack of traceability does not imply that a MCCB is improper. Furthermore, lack of traceability for MCCBs ordered more than five years ago is likely due to the unavailability of records for these type of components not because of the involvement of suspect suppliers.

Our review of industry Bulletin 88-10 responses concludes that the expanded Bulletin requirements contained in the proposed supplement are not warranted. It appears the Staff is concerned about the shortcomings of relatively few utility responses. We believe it more appropriate for the NRC Staff to address the few deficient responses directly with individual utilities rather than issue new generic requirements on the entire industry.

We request you carefully consider these and our specific comments contained in Attachment 1, particularly those addressing items 1, 3 and 4 of the draft supplement. We are willing to meet with NRC Staff to further articulate these concerns.

We think it important that NUMARC and NRC continue to work together on the MCCB issue. Please call Alex Marion or Russell Bell at any time for further discussion of our questions on the Bulletin supplement or the attached comments.

Sincerely,

William H. Rasin

William H. Rasin
Director, Technical Division

Attachment
WHR/sal

Attachment 1

NUMARC COMMENTS ON
BULLETIN 88-10, SUPPLEMENT 1 (DRAFT)

Item 1. The language of Bulletin 88-10 reflects the important understanding reached last year between NUMARC and NRC Staff that replacement MCCBs are of primary concern, not MCCBs ordered as part of original plant equipment. The Bulletin appropriately states,

> "This (original) equipment appears to have been procured during plant construction from CBMs with full certification. The large quantities of electrical assemblies or components procured under bid packages during plant construction reduce the possibility of any original plant equipment being supplied by vendors doing refurbishing."

With this language, the Staff acknowledged the low likelihood that MCCBs procured during plant construction were supplied by refurbishers and concluded, "concerns addressed in this Bulletin do not apply to (original equipment)." The supplement reverses this position by requiring original MCCBs to meet fully the traceability requirements of Bulletin 88-10.

We find the original position of the Staff expressed in the Bulletin to be sound, and we find it inappropriate for the Staff to expand the scope of Bulletin 88-10.

We suggest deletion of item 1 in its entirety.

Item 2. Delete the second sentence. The first sentence appropriately addresses the Staff's concern regarding unacceptability of visual inspection by itself as an alternative to verifiable traceability.

Item 3. This item represents a clear expansion of Bulletin 88-10 scope which is inappropriate at this time. The Bulletin states, "...a Supplement may be issued to include...a longer procurement review period if warranted by the results of...this Bulletin." Five years was selected as a reasonable period of review by the Staff due to:

1) likelihood that records would be available to establish traceability, and

2) evidence that MCCB refurbishment was more prevalent in the last five years.

The five year review of industry MCCBs required by Bulletin 88-10 is not yet complete, and results to date do not suggest the need for additional review.

However, independent of the requirements of Bulletin 88-10, we believe that utilities would continue investigating installed MCCBs associated with <u>clearly suspect</u> MCCBs (e.g. those from a known suspect supplier or exhibiting signs of refurbishment). Lack of traceability, in our view, does not by itself provide sufficient cause for expanded actions.

traceability, in our view, does not by itself provide sufficient cause for expanded actions.

We are greatly concerned by the implications of supplement item 3 when viewed in conjunction with item 1. The result of these two new requirements is that a utility which cannot establish verifiable traceability per Bulletin 88-10 for original equipment MCCB spares, may, by association, be required to consider as non-traceable large numbers (perhaps all) originally installed MCCBs.

Based on our review of Bulletin responses, an increase in the scope of Bulletin 88-10 is not justified at this time. We believe, therefore, that supplement item 3 should be deleted completely.

Item 4. The first sentence of item 4 offers a useful clarification of Bulletin 88-10. We suggest adding the phrase "neither supplied by known refurbishers nor exhibiting evidence of refurbishment" after "CBs" in the first line. Also, after "Bulletin 88-10" in line 3, the phrase "or equivalent manufacturer recommended evaluation/test program" should be inserted.

However, we must caution against reliance placed on the test program detailed in Attachment 1 of the Bulletin given the positions of the National Electrical Manufacturers Association and Underwriters Laboratories regarding the inadequacy of such testing.

We feel the second sentence of item 4 is inappropriate and should be deleted.

Item 5. We find this item appropriate and suggest the first sentence be rewritten as follows, "Regardless of the number of CBs stored..."

Item 6. The Staff has acknowledged that traceability of MCCBs to corporate facilities of the CBM such as controlled warehouses, distribution centers or production facilities is sufficient for the purpose of Bulletin 88-10. We therefore suggest the following replacement language for item 6:

> "CBs demonstrated traceable to the CBM by test report, certificate of compliance, purchase documents, shipping records or other procurement records are acceptable for safety-related service. The CBM is considered to include all corporate controlled facilities including controlled warehouses, distribution centers and production facilities.
>
> Specifically, a certificate of compliance from other than the CBM (e.g. equipment supplier) would not constitute adequate traceability of MCCBs without such procurement records as suggested above demonstrating the traceability of each individual procurement to the CBM. Traceability should be demonstrated by documents in the utility's possession or verified by audit or other appropriate means. Telephone discussions with vendors are not acceptable basis for establishing traceability."

National Electrical Manufacturers Association
2101 L Street, N.W. Suite 300
Washington, D.C. 20037-1581 (202) 457-8400
Telex 904077 NEMA WSH

May 16, 1989

Mr. William Brach
Chief of Vendor Inspection Branch
Division of Reactor Inspection and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: May 11, 1989 Draft of NRC Bulletin
88-10, Supplement 1

Dear Mr. Brach:

Regarding Item 4 of the subject draft supplement, we would like to reaffirm the points that we made to you in our October 24, 1988 and December 22, 1988 letters to you.

Neither the bulletin 88-10 tests nor visual inspection alone, or in combination, are adequate to ensure breaker performance. Any breaker that cannot be traced to the original circuit breaker manufacturer should be removed from service for inspection and test. The manufacturer of the breaker in question should be contacted for specific recommendations.

We appreciate the opportunity to comment on this draft supplement and continue to stand ready to work with NRC in bringing this issue to a speedy resolution.

If you have any questions regarding our comments, please feel free to give me a call.

Sincerely,

Robert W. Baird
Division Staff Executive

cc: J. Bhatia, UL
U. Potapovs, NRC
A. Marion, NUMARC
T. F. on Rebuilt Breakers

1p

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

[illegible] 1988

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

FROM: James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: DRAFT BULLETIN REGARDING MOLDED-CASE CIRCUIT BREAKERS

The Office of Nuclear Reactor Regulation (NRR) previously requested by memorandum dated August 19, 1988, that the Committee to Review Generic Requirements (CRGR) consider a proposed bulletin regarding nonconforming electrical equipment, components and devices. The CRGR approved the issuance of the bulletin at the August 24, 1988, meeting. The draft bulletin was not issued and has been substantially revised since then; consequently, NRR requests that the CRGR consider the revised proposed bulletin.

The revised proposed bulletin requests that holders of operating licenses and construction permits take certain actions to provide reasonable assurance that molded-case circuit breakers (CBs) purchased for safety-grade applications without traceability to the original circuit breaker manufacturer (CBM) perform their safety functions. The proposed bulletin also requires that addressees provide reports listing those molded-case CBs that could not be traced to the CBM as well as the results of tests performed in accordance with the actions requested in the bulletin.

The revised proposed bulletin and the information required to support the issuance of this bulletin are enclosed. Lawrence Shao, Director, Division of Engineering and Systems Technology, is the sponsoring Division Director.

James H Sniezek

James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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X20811

J. Guillen, OGCB/DOEA/NRR
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8811180083

Distribution:
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A. Thadani
L. Shao
C. E. Rossi
B. Grimes
OGC
J. Sniezek
F. Miraglia
JGuillen

Enclosure

CRGR Item IV.B. Contents of Packages Submitted to CRGR
(Rev. 4, Stello to List 042387, dcs 41860 342 ff)

The following requirements apply for proposals to reduce existing requirements or (regulatory) positions as well as proposals to increase requirements or (regulatory) positions. Each package submitted to the CRGR for review shall include fifteen (15) copies of the following information:

SUBJECT: BULLETIN REGARDING NONCONFORMING MOLDED-CASE CIRCUIT BREAKERS

Question:

I. The proposed generic requirement as it is proposed to be sent out to licensees.

Response:

The proposed generic requirement is spelled out in the proposed bulletin.

Question:

II. Draft staff papers or other underlying staff documents supporting the requirements or staff (regulatory) positions. (A copy of all materials referenced in the document shall be made available upon request to the CRGR staff. Any committee member may request CRGR staff to obtain a copy of any referenced material for his or her use.)

Response:

A. Proposed NRC Bulletin No. 88-XX: "Nonconforming Molded-Case Circuit Breakers."

B. Previous proposed NRC Bulletin No. 88-xx: "Nonconforming Electrical Equipment Components and Devices."

C. Letter from Underwriters Laboratories, Inc., dated October 25, 1988.

D. Letter from General Electric Company (GE), dated October 28, 1988.

E. Letter from National Electric Manufacturers Association (NEMA), dated October 24, 1988.

Question:

III. Each proposed requirement or staff (regulatory) position shall contain the sponsoring office's position as to whether the proposal would increase requirements or staff (regulatory) positions, implement existing requirements or staff (regulatory) positions, or would relax or reduce existing requirements or staff (regulatory) positions.

Response:

A. The proposed bulletin requires verification that existing requirements are being met.

B. Staff regulatory positions are not altered by this proposed bulletin. Addressees are only being requested to review their records, provide the NRC with the requested information, and to comply with existing regulatory requirements.

Question:

IV. The proposed method of implementation along with the concurrence (and any comments) of OGC on the method proposed.

Response:

The staff proposes to promulgate this proposed requirement by means of a bulletin. This method has been effective in the past. OGC reviewed the previous proposed bulletin and had no legal objection to that package, including the proposed bulletin, and all of their comments were incorporated. None of the subsequent changes necessitated OGC's review.

Question:

V. Regulatory analyses generally conforming to the directives and guidance of NUREG/BR-0058 and NUREG/CR-3568. (Make sufficient to address the Paper-work Reduction Act, the Regulatory Flexibility Act and Executive Order 12291).

Response:

A. This request for information was approved by the Office of Management and Budget under blanket clearance number 3150-0011 as meeting the requirements of the Paper Reduction Act and Executive Order 12291.

B. Since this is not a rulemaking action, the Regulatory Flexibility Act does not apply.

Question:

VI. Identification of the category of reactor plants to which the generic requirement or staff position is to apply (that is, whether it is to apply to new plants only, new OLs only, OLs after a certain date, OLs before a certain date, all OLs, all plants under construction, all plants, all water reactors, all PWRs only, some vendor types, some vintage types such as BWR 6 and 4, jet pump and nonjet pump plants, etc.)

Response:

The proposed requirements apply to all holders of operating licenses and construction permits for nuclear reactors.

Question:

VII. For each such category of reactor plants, an evaluation which demonstrates how the action should be prioritized and scheduled in light of other congoing regulatory activities. The evaluation shall document for consideration information available concerning any of the following factors as may be appropriate and any other information relevant and material to the proposed action:

Response:

Response to this item is not required pursuant to Revision 4 of the CRGR Charter, Section III.D., since the requirements of the proposed bulletin are intended to provide the NRC with information and will bring licensees into compliance with existing regulatory requirements.

Question:

VIII. For each evaluation conducted pursuant to 10 CFR 50.109, the proposing office director's determination, together with the rationale for the determination based on the considerations of all the above, that:

A. There is a substantial increase in the overall protection of public health and safety or the common defense and security to be derived from the proposal; and

Response:

The specific actions in the bulletin will address nonconformances and concerns identified by the staff during recent inspections and provide a reasonable assurance that these CBs perform their intended functions. The proposed bulletin also requests addressees to perform certain actions and provide certain information regarding molded-case CBs to assess compliance with existing regulatory requirements. The information will be selectively audited by the staff to identify possible additional generic concerns.

Question:

B. The direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection.

Response:

The actual implementation costs will vary between 1,000 to 10,000 man-hours for each addressee depending on the total number of molded-case CBs in stores for or installed in safety-grade applications, the number of non-traceable CBs identified, the failure rate of the CBs tested, and replacement costs. These estimates are highly speculative due to the unknown scope of the problem and the adequacy of addresses' records.

It is estimated that a typical holder of an operating license with computerized procurement records will require approximately 2,000 man-hours (or $120,000 at $60 per man-hour) to meet the bulletin requirements and will spend approximately $30,000 to replace CBs that fail the tests included in the bulletin. These estimates are calculated based on the licensee having 300 molded-case CBs in stores for safety-grade application and an additional 300 that have been installed between August 1, 1983 and August 1, 1988, in safety-grade applications. In addition, 25% of the breakers are assumed not to be traceable to the CBM and a total of 30 are assumed to fail the tests included in the bulletin and require replacement. Replacement breakers are estimated to cost approximately $1,000 each.

It is estimated that typical addressees that do not have computerized procurement records will require approximately 8,000 man-hours (or $480,000 at $60 per man-hour) to implement the bulletin requirements and spend $30,000 to replace CBs that fail the tests included in the bulletin. These estimates are based on the same assumptions as the example above, except that search and traceability determination may take an additional 6,000 man-hours.

Actions that can not be completed during normal operation may be completed during the next refueling outage beginning after March 1, 1988; therefore, plant shutdowns are not required. In addition, addressees that cannot meet the bulletin schedules can justify to the NRC their proposed alternative schedule.

The NRC staff feels that the costs of implementation of the bulletin are justified in view of the increased confidence in the safety of nuclear power reactors.

Question:

IX. For each evaluation conducted for proposed relaxations or decreases in current requirements of staff positions, the proposing office director's determination, together with the rationale for the determination based on the considerations of the above, that the public health and safety and the common defense and security would be adequately protected if the proposed reduction in requirements or (regulatory) positions were implemented, and the cost savings attributed to the action would be substantial enough to justify taking the action.

Response:

This item is not applicable to the proposed bulletin because no relaxation or decrease in current requirements is being proposed.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

DRAFT

November xx, 1988

NRC Bulletin No. 88-xx: NONCONFORMING MOLDED-CASE CIRCUIT BREAKERS

Addressees:

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose:

The purpose of this bulletin is to request that addressees take actions to provide reasonable assurance that molded-case circuit breakers (CBs[1]), including CBs used with motor controllers, purchased for use in safety-grade[1] applications without verifiable traceability[1] to the circuit breaker manufacturer (CBM)[1] perform their safety functions.

Description of Circumstances:

NRC Information Notice No. 88-46, "Licensee Report of Defective Refurbished Circuit Breakers," dated July 8, 1988 and Supplement 1 thereto, dated July 21, 1988, discussed a report by Pacific Gas and Electric Company that indicated that its Diablo Canyon Nuclear Power Plant was supplied 30 CBs by Anti-Theft Systems, Inc. through a local electrical distributor. These CBs (Square D molded-case, type KHL 36125) were intended for use in non-safety-grade applications at the Diablo Canyon Nuclear Power Plant. Square D Company reported that an inspection and testing of these breakers determined that the CBs were refurbished Square D Company equipment. Furthermore, Square D reported that several of the circuit breakers tested did not comply with Square D or Underwriters Laboratories, Inc. (UL) specifications for all of the electrical tests performed. Information Notice No. 88-46 also listed several California companies that were involved in supplying surplus refurbished and possibly defective refurbished electrical equipment to the nuclear industry.

During recent NRC inspections, additional examples were identified that indicate a potential safety concern regarding electrical equipment supplied to nuclear power plants. The NRC is concerned that equipment being procured as new and assumed to meet all applicable plant design requirements and/or original manufacturer's specifications may, in fact, not conform to these requirements and specifications.

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1. Refer to Attachment 2 for Definition of Terms

The actions requested in this bulletin are limited to molded-case CBs. Molded-case CBs are tested and calibrated at the manufacturer's plant in accordance with recognized industry standards, such as UL-489, "Molded Case Circuit Breakers and Circuit Breaker Enclosures," and National Electrical Manufacturers Association (NEMA)-AB1, "Molded-Case Circuit Breakers." Since molded-case CBs have factory-calibrated and sealed elements, any unauthorized modification or refurbishing of these CBs could jeopardize their design capability and reliability.

The NRC is concerned that the reliability and capabilities of refurbished CBs purchased as commercial grade (non-Class 1E) for later upgrading to safety-grade (Class 1E) applications may not meet the minimum commercial grade standards. In addition, the NRC is concerned about the reliability and capabilities of commercial grade breakers upgraded to safety-grade because of some observed inadequacies in the dedication process and numerous failures found during the testing of some of these breakers. In order to properly dedicate electrical items procured as commercial grade for subsequent use in safety-grade applications, the dedication process should build from the commercial grade quality, include a proper evaluation of seismic and environmental qualification, confirm critical parameters, and include testing as appropriate.

Safety-grade electrical equipment purchased as Class 1E from the CBM, or corporate divisions associated with the CBM, is of lesser concern as this equipment is controlled under quality assurance (QA) programs that conform to Appendix B of 10 CFR Part 50. The controls imposed by these QA programs are more stringent than those exercised in the manufacturing of commercial grade equipment. While the upgrading programs of CBMs, or corporate divisions associated with the CBMs, may vary in quality, the controls exercised over the procurement and manufacturing activities provide reasonable assurance that improperly refurbished components have not been introduced and passed through the upgrading process. Furthermore, the redundancy of safety systems and the in-service use of these components prov'de a reasonable basis for accepting installed replacement components tha' ve been procured as safety-grade from the CBM, or from corporate divisions associated with the CBM.

The NRC currently believes that the concerns addressed in this bulletin do not apply to electrical equipment (safety-grade and commercial grade) originally installed in plants. This equipment appears to have been procured during plant construction from CBMs with full certification. The large quantities of electrical assemblies or components procured under bid packages during plant construction reduce the possibility of any original plant equipment being supplied by vendors doing refurbishing.

The NRC expects all addressees to participate in a joint industry program that ensures that non-safety-grade molded-case CBs, which may have been installed as replacements, installed during modifications, or are being maintained as stored spares and that were not procured from the CBM, or whose original source has not been determined, are suitable for their intended service. A joint industry report describing the program is expected within 180 days of receipt of this bulletin.

The NRC requested and received comments from the Nuclear Management and Resources Council (NUMARC), the National Electrical Manufacturers Association (NEMA), and the Underwriters Laboratories, Inc. (UL), during the preparation of this bulletin. These comments were considered and some were appropriately incorporated into this bulletin.

NEMA has commented to the NRC that determination of the critical performance characteristics of durability and short-circuit capabilities of circuit breakers requires destructive testing of selected breakers that are representative of breakers to be placed in service. Because a refurbished breaker may not have been refurbished under controlled conditions to conform to a proven design, destructively testing selected breakers will not infer anything about a refurbished breaker. UL provided specific comments on the tests in Attachment 1 of this bulletin. In addition, they stated that, "it is UL's opinion that the test program is not adequate to provide assurance that the tested, non-traceable, circuit breakers would be suitable for their intended purpose." The NRC agrees with these comments. The non-destructive testing in Attachment 1 of this bulletin, however, is directed at ensuring that the circuit breakers will perform those functions most important to ensuring reactor safety even though the tests will not verify the capability of performing certain functions that can only be verified by destructive testing.

The NRC investigation of this issue is not complete. A supplement to this bulletin may be issued to include other electrical equipment or a longer procurement review period if warranted by the results of the ongoing evaluations or the results of testing requested in this bulletin.

Actions Requested:

1. All addressees are requested to perform the following review by March 1, 1989:

   a. Identify all molded-case CBs purchased prior to August 1, 1988, that are being maintained as stored spares for safety-grade (Class 1E) applications or commercial grade CBs that are being maintained as stored spares for future use in safety-grade applications; this includes CBs purchased from a CBM or from any other source. If the number of these stored spare CBs is less than 50 at a nuclear plant site, then randomly select CBs purchased between August 1, 1983 and August 1, 1988 that have been installed in safety-grade applications as replacements or modifications to form a minimum sample of 50 CBs.

   b. Verify the traceability of these CBs.

   c. Identify the number, manufacturer, model number, and to the extent possible the procurement chain for all CBs that cannot be traced to the CBM. For installed CBs, also identify each system in which they are/were installed.

2. All holders of operating licenses who identify installed CBs per item 1 above that cannot be traced to a CBM are requested to prepare, within 30 days of the completion of item 1 above, an analysis justifying continued operation until items 1 through 5 of the actions requested in this bulletin have been completed.

3. All addressees who identify 80 percent or more CBs traceable to the CBM per item 1 above are requested to test the CBs that are not traceable to the CBM in accordance with the test program described in Attachment 1. Any installed CBs that fail any of these tests should be replaced with components that meet the criteria of item 7 of the actions requested or that pass all tests in accordance with the testing program described in Attachment 1. If more than 10 percent of the CBs tested fail any of the tests described in Attachment 1, continue with item 4; otherwise, proceed to item 6 of the actions requested.

   Holders of operating licenses are requested to complete this testing program before startup from the first refueling outage beginning after March 1, 1989. Holders of construction permits are requested to complete this testing program before fuel load.

4. All addressees who identify less than 80 percent of the CBs traceable to the CBM per item 1 above or who identify a failure rate of more than 10 percent for the CBs tested per item 3 above are requested to perform the following actions:

   a. Identify all molded-case CBs that have been purchased between August 1, 1983 and August 1, 1988, and installed in safety-grade applications as replacements or installed during modifications.

   b. Verify the traceability of these CBs.

   c. Identify the number, manufacturer, model number, system in which they are/were installed, and to the extent possible the procurement chain for all those CBs that cannot be traced to the CBM.

5. All addressees who identify installed CBs that cannot be traced to the CBM per item 4 above are requested to replace these CBs with components that meet the criteria of item 7 of the actions requested or to test them in accordance with the program described in Attachment 1; CBs that fail any of these tests should be replaced with components that meet the criteria of item 7 of the actions requested or that pass all tests in accordance with the test program described in Attachment 1.

Holders of operating licenses are requested to replace or to test at least one-half, or all if the total number is less than 75, of these installed CBs before startup from the first refueling outage beginning after March 1, 1989. The remaining breakers should be replaced or tested before startup from the second refueling outage beginning after March 1, 1989.

Holders of construction permits are requested to replace or to test these installed breakers before fuel load.

6. Information generated while performing the actions requested in items 1, 2, 3, 4, and 5 above should be documented and maintained for possible NRC audit for a period of 5 years after the completion of all requested actions.

7. With the exception of actions taken in response to items 3 and 5 of the actions requested above, molded-case CBs installed in safety-grade applications after August 1, 1988 should be:

   a. Manufactured by and procured from a CBM under a 10 CFR 50, Appendix B, program; or

   b. Procured from a CBM or others with verifiable traceability to the CBM, in compliance with applicable industry standards, and upgraded to safety-grade by the licensee or others using an acceptable dedication program. Tests equivalent to those in Attachment 1 are acceptable for a dedication process of CBs traceable to the CBM. In addition, seismic and environmental qualification requirements should be addressed by additional testing or analysis based on plant-specific considerations.

8. Addressees that cannot meet the schedule for the actions requested above and/or the corresponding reporting requirements below, should justify to the NRC their proposed alternative schedule.

Reporting Requirements:

1. All holders of operating licenses are required to provide a written report by April 1, 1989, that:

   a. Confirms that only molded-case CBs that meet the criteria of item 7 of the actions requested are being maintained as stored spares for future use in safety-grade applications.

   b. Summarizes the total number, manufacturer, model number, and to the extent possible the procurement chain of those CBs that could not be traced to the CBM in items 1 and 4 of the actions requested. For installed CBs, also identify each system in which they are/were installed. If item 4 of the actions requested has not been completed by April 1, 1989, due to the schedule for tests in item 3 of the actions requested, this information should be updated within 30 days of the completion of item 4 to address those additional CBs that could not be traced to the CBM.

c. Confirms that items 1, 2, 3, 4, 5, 6 and 7 of the actions requested have been completed or will be implemented as requested.

2. All holders of operating licenses are required to submit a report that summarizes available results of tests conducted in accordance with items 3 and 5 of the actions requested within 30 days after startup from the first and second refueling outages beginning after March 1, 1989. These reports should include the number, manufacturer, model number, and to the extent possible the procurement chain of CBs tested. For CBs that fail these test(s), these reports should indicate the test(s) and the values of test parameter(s) at which they failed.

3. All holders of construction permits are required to provide a written report by April 1, 1989, that:

   a. Confirms that only molded-case CBs that meet the criteria of item 7 of the actions requested are being maintained as stored spares for future use in safety-grade applications.

   b. Summarizes the total number, manufacturer, model number, and to the extent possible the procurement chain of those CBs that could not be traced to the CBM in items 1 and 4 of the actions requested. For installed CBs, also identify each system in which they are/were installed. If item 4 of the actions requested has not been completed by April 1, 1989, due to the schedule for tests in item 3 of the actions requested, this information should be updated within 30 days of the completion of item 4 to address those additional CBs that could not be traced to the CBM.

   c. Confirms that items 1, 3, 4, 5, 6 and 7 have been completed or will be implemented before fuel load.

4. All holders of construction permits are required to submit a report that summarizes the results of tests conducted in accordance with items 3 and 5 of the actions requested within 30 days after fuel load. The report should include the number, manufacturer, and model number of all breakers tested. For CBs that fail these test(s), the report should indicate the test(s) and the values of test parameter(s) at which they failed.

The written reports required above shall be addressed to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended. In addition, a copy shall be submitted to the appropriate Regional Administrator.

This request is covered by Office of Management and Budget Clearance Number 3150-0011 which expires December 31, 1989. The estimated burden hour is 1000 to 10,000 man-hours per plant response, including assessment of these requirements, searching data sources, testing, and analyzing the data, and preparing the required reports. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to the Office of Management and Budget, Room 3208, New Executive Office Building, Washington, D.C., 20503, and to the U.S. Nuclear Regulatory Commission, Records and Reports Management Branch, Office of Administration and Resource Management, Washington, D.C., 20555.

If you have any questions regarding this matter, please contact one of the technical contacts listed below or the Regional Administrator of the appropriate NRC regional office.

DRAFT

Charles E. Rossi, Director
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Office of Nuclear Reactor Regulation

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Attachments:
1. Test Program for Molded Case Circuit Breakers
2. Definition of Terms
3. List of Recently Issued NRC Bulletins

Attachment 1
NRCB 88-xx
November xx, 1988

TEST PROGRAM FOR MOLDED-CASE CIRCUIT BREAKERS

DRAFT

1.0 Test Program Objectives

The objective of this proposed test program is to verify the reliability and capabilities of molded-case circuit breakers (CBs).

For the safety of personnel and others involved with the activities related to these proposed tests, appropriate safety practices, such as ANSI/NFPA 70E, "Electrical Safety Requirements for Employee Workplaces," Part II, should be followed.

These proposed tests have been based on tests described in industry standards, such as NEMA AB-1, "Molded-Case Circuit Breakers," NEMA AB-2, "Procedures for Field Inspection and Performance Verification of Molded-Case Circuit Breakers Used in Commercial and Industrial Applications," UL 489 "Molded Case Circuit Breakers and Circuit Breaker Enclosures," and NETA STD ATS-1987, "National Electrical Testing Association, Acceptance Testing Specifications."

2.0 Test Procedures for CBs

The following tests should be performed in the sequence listed. CBs failing any of these tests should be considered unacceptable for safety-grade applications.

2.1 Mechanical Test

The CB should be operated, reset, and closed a minimum of five times, to ensure that the latching surfaces are free of any binding.

2.2 Individual Pole Resistance or Millivolt Drop Test (Ref. NETA STD ATS-1987 & NEMA AB-2)

The contact resistance of each pole of the CB should be measured at ambient temperature. Three readings of each pole should be taken with the CB operated without load between each reading. The average of three readings for each pole should be calculated and compared with the manufacturer's contact resistance data or with those values of similar CBs from the same manufacturer. Also, the average value for each pole should be compared with the average of the other poles and the difference between the pole values should not exceed 50 percent of the lowest value; or

A millivolt drop test may be performed by applying a direct current across the closed CB contacts and measuring the voltage drop due to the contact resistance. The millivolt drop test should be performed at room temperature. Direct current should be applied across each pole and the millivolt drop and test current recorded for each pole. Three readings of each pole should be taken with the CB operated without load between each reading. The average of the three readings for each pole should be calculated and compared with the manufacturer's value for acceptance of the breaker.

2.3 Rated Current Hold-In Test (Ref. NEMA AB-1 & UL 489)

This test should be conducted at 100% rated current and at an ambient air temperature of 25°C ± 3°C, and followed by a test at 135% rated current and at an ambient temperature of 25°C ± 3°C.

Equal 100% rated currents should be applied to all poles of the CB. The CB must not trip within 1 hour for CBs rated 50 amperes or below or within 2 hours for CBs rated over 50 amperes during this test. At the end of the 100% rated current test, the current should be increased to 135% and the CB should trip within 1 hour for CBs rated 50 amperes or below or within 2 hours for CBs rated over 50 amperes.

2.4 Overload Test (Ref. NEMA AB-1 & UL 489)

This test consists of one operating cycle (i.e., closing action followed by an opening action) of the CB at 600% rated current. This test may be conducted at low voltage. There should be no electrical or mechanical breakdown of the CB during this test.

2.5 Instantaneous Trip Test (Ref. NEMA AB-1 & UL 489)

2.5.1 Fixed Instantaneous Setting CBs

Each pole of the CB should be tested for pickup of the instantaneous unit. Each pole must be between 75% and 125% of the instantaneous trip rating. The trip time should not exceed 0.1 seconds (6 cycles).

2.5.2 Adjustable Instantaneous Setting CBs

This test is the same as that in Section 2.5.1 except that each pole must be tested at the lowest and highest settings.

The trip value for the lowest setting should be between 75% and 125% of the lowest setting, and the highest setting should be between 80% and 120% of the highest setting.

2.5.3 Short-Time Trip Setting Test

This test is applicable only if the CB is equipped with the short-time delay trip. This test should be conducted at an ambient air temperature of 25°C ± 3°C. The operation of the short-time delay unit should be within 90% and 125% of the overcurrent setting of the CB as shown on the manufacturer's time-current curves.

2.6 Time Delay Overcurrent Trip (Ref. NEMA AB-2)

This test should be conducted at an ambient air temperature of 25°C ± 3°C.

A current of 300% (at low voltage) of the marked rating should be applied to each pole of the CB. The trip time for each pole should be compared with the time shown in the CB manufacturer's time-current curves. If the test trip times obtained for each pole are not within the time band shown on the CB manufacturer's time-current curves, then the test trip must not exceed the time specified in Table 1 and the acceptance of the CBs must be evaluated with the criteria listed below:

TABLE 1
VALUES FOR OVERCURRENT TRIP TEST
(AT 300% OF RATED CONTINUOUS CURRENT OF CIRCUIT BREAKER)
(REF. NEMA AB-2)

Breaker Voltage Volts	Range of Rated Continuous Current Amperes	Maximum Tripping Time In Seconds
240	15-45	50
240	50-100	70
600	15-45	70
600	50-100	125
240	110-225	200
240	250-400	300
600	110-225	250
600	250-400	300
600	450-600	350
600	700-1200	500
600	1400-2500	600
600	3000-5000	650

Minimum Tripping Time: If the minimum tripping times are lower than indicated by the manufacturer's time-current curves for the CB under test, the CB should be retested after it has been cooled to 25°C. If the values obtained are still lower after retest, the coordination with upstream and downstream CB should be evaluated. If no problem with coordination is indicated, then the CB is acceptable.

Maximum Tripping Time: If the tripping time exceeds the maximum tripping time shown on the manufacturer's time-current curves but is below the time shown in Table 1, check the CB time against the protection requirements of the circuit (such as cable, penetration, etc.) to ensure that the CB provides the protection, as well as the coordination with upstream and downstream CBs. If the CB provides the necessary protection and coordination, then the CB is acceptable.

Maximum Allowable Time: If the tripping time of the CB exceeds the trip time shown in Table 1, the breaker is unacceptable for Class 1E applications.

2.7 Dielectric Tests (Ref. NEMA AB-1 & UL-489)

The dielectric test should be conducted at an ac test voltage of 1760 volts (80% x [2 x rated voltage + 1000 volts]), or at 2500 volts dc for 1 minute withstand. The dielectric test should be conducted for (1) line to load terminals with CB open, (2) line to line terminals with CB closed, and (3) pole to ground with CB open, and (4) pole to ground with CB closed.

DEFINITION OF TERMS

CIRCUIT BREAKER MANUFACTURER (CBM)

The manufacturing facility that actually produced the circuit breaker being purchased.

VERIFIABLE TRACEABILITY

Documented evidence such as a certificate of compliance that establishes traceability of purchased equipment to the CBM. If the certificate of compliance is provided by any party other than the CBM, the validity of such certificate must be verified by the licensee or permit holder through an audit or other appropriate means.

DEDICATION PROCESS

The process by which commercial grade (non-Class 1E) equipment is upgraded to safety-grade (Class 1E) and is thereby considered qualified for use in nuclear safety-related applications. The dedication process must include:

a. A technical evaluation to determine the characteristics critical to fulfilling the safety function(s).

b. An acceptance process to ensure that those critical characteristics are met.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON D.C. 20555

AUGUST XX, 1988

NRC BULLETIN 88-XX: NONCONFORMING ELECTRICAL EQUIPMENT, COMPONENTS AND DEVICES

ADDRESSEES:

All holders of operating licenses or construction permits for nuclear power reactors.

PURPOSE:

The purpose of this bulletin is to request that addressees take actions to assure that installed molded case circuit breakers (CBs) and motor controllers (MCs) comply with plant design requirements such as IEEE, Underwriters Laboratory (UL), National Electrical Manufacturers Association (NEMA) or original manufacturer's specifications. The scope of requested actions are limited to molded case CBs and MCs, procured as replacements for original plant equipment, that have or may have been refurbished.

DESCRIPTION OF CIRCUMSTANCES:

NRC Information Notice No. 88-46 "Licensee Report of Defective Refurbished Circuit Breakers," dated July 8, 1988 and its supplement 1, dated July 21, 1988, discussed a report by Pacific Gas and Electric which indicated that the Diablo Canyon Nuclear Power Plant had been supplied 30 CBs by Anti-Theft Systems, Inc. through a local electrical distributor. These CBs (Square D molded case, type KHL 36125), were intended for use in non-safety-related applications at the Diablo Canyon nuclear power plant. Square D company reported that the inspection and testing of these breakers determined that the CBs were refurbished Square D Company equipment. Further, Square D reported that many of the circuit breakers tested did not comply with Square D or UL specifications for all of the electrical tests performed. Information Notice No. 88-46 also listed several California companies that were involved in supplying possibly defective refurbished electrical equipment to the nuclear industry.

NRC Information Notice No. 88-19 "Questionable Certification of Class 1E Components" dated April 26, 1988 discussed a 10 CFR Part 21 notification, submitted on April 1, 1988 by the Wolf Creek Nuclear Operating Corporation (WCNOC), which brought into question the validity of Certificates of Compliance issued by Planned Maintenance Systems, Inc. (PMS) for Class 1E fuses. In response to this notification, the NRC staff inspected PMS. Components supplied by PMS with questionable certification included circuit breakers, fuses and relays. The NRC investigation of this issue is not complete and, if warranted, a supplement to this bulletin may be issued.

These examples indicate a potential safety concern regarding electrical equipment supplied to nuclear power plants. The NRC is concerned that the equipment procured as being new and assumed to meet all applicable plant design requirements and/or original manufacturer's specifications may in fact not conform to these requirements and specifications.

The actions requested in this bulletin are only related to molded case CBs and MCs due to their widespread use and applications and the potential impact of their failure. Molded case CBs are tested and calibrated at the manufacturer's plant in accordance with recognized industry standards (UL-489-Molded-Case Circuit Breakers and Circuit Breaker Enclosures, NEMA-AB1-Molded-Case Circuit Breakers). Since molded case CBs have factory-calibrated and sealed elements, any unauthorized modification or refurbishing of the CBs jeopardize their capability and reliability, as well as the manufacturer's warranty. The MCs are also built in accordance with similar NEMA standards and are UL listed, and any unauthorized refurbishing of such equipment compromises its integrity. Therefore, reliable operation of refurbished molded case CBs and MCs installed in nuclear power plants cannot be relied upon, due to lack of assurance of uniformity of parts, materials and workmanship used in unauthorized refurbishing activities.

The NRC is concerned about refurbished CBs and MCs purchased as commercial grade (non-Class 1E) for later upgrading to safety grade (Class 1E) applications, because these CBs and MCs may not meet the minimum commercial grade standards. In order to dedicate electrical items procured as commercial grade and subsequently used in safety-related systems, the dedication process should build from the commercial grade quality, and include proper evaluation of seismic and environmental qualification, as well as confirmation of critical operating parameters and functional testing as appropriate. When refurbished CBs and MCs are upgraded to safety grade applications, the NRC is concerned that the licensee's normal dedication process may not be adequate for properly evaluating the acceptability of the components.

The safety grade electrical equipment originally purchased as Class 1E is not a concern, since this equipment is purchased and installed under quality assurance (QA) programs which conform with Appendix B of 10 CFR 50. The controls imposed by these QA programs are more stringent than the controls exercised in procurement of commercial grade equipment. Further, the requirements of these QA programs are well established and known to electrical equipment suppliers and are subject to frequent audits.

The NRC believes that the safety concern does not extend to the electrical equipment (Class 1E and non-Class 1E) originally installed since this equipment was procured during plant construction from original manufacturers with full certification. Moreover, this equipment was inspected and tested before the nuclear plant became operational. The large quantities of electrical assemblies or components procured under bid packages minimizes the possibility of small vendors doing refurbishing business having supplied original equipment.

The actions requested in this bulletin apply only to molded case CBs and MCs procured as replacements for original plant equipment as indicated in action item 1 below. However, the NRC investigation of this issue is not complete. A supplement to the bulletin may be issued to include other electrical equipment if warranted by the results of the ongoing evaluation.

ACTIONS REQUESTED:

1. All addressees are requested to review their records to identify the number, types and applications of installed (or stored spare) replacement molded case CBs and MCs, that were procured as commercial grade (non-Class 1E) and later upgraded to safety grade (Class 1E), that were not procured from the original manufacturers,[1] or whose procurement source has not been determined. Replacement CBs and MCs, procured from distributors who cannot demonstrate the equipment was procured directly from the original equipment manufacturer without intermediate refurbishment, must be assumed to be unacceptable.

2. All addressees that identified installed CBs or MCs per item 1 above are requested to replace these CBs or MCs with fully qualified components, or provide justification for continued operations (JCO) until the suspect CBs or MCs are replaced, to be completed not later than before startup after the second refueling outage from the date of this bulletin. (for all holders of CPs, the CB or MC replacement should be completed prior to fuel load)

3. All addressees that identified stored spare CBs or MCs per item 1 above are requested to take appropriate actions to ensure that these CBs are not used for safety-related service.

1. Original Manufacturers are defined as those companies that manufacture the CBs. Supply companies distributing CBs (WESCO, GESCO, GRAYBAR and others) are not considered as original manufacturers of CBs.

4. All addressees are requested to develop a program for the installed or in storage non-safety related replacement molded case CBs and MCs, that were not procured from the original manufacturer or whose original source has not been determined, to assure that they are suitable for the intended service. A joint industry program which attains this objective is encouraged.

5. Information generated during the completion of items 1,2,3 and 4 above shall be documented and maintained for possible NRC audit.

REPORTING REQUIREMENTS:

1. All holders of operating licenses for nuclear power reactors are required within 120 days of the receipt of this bulletin to provide a written report that:

   a. confirms that no molded case CBs or MCs have been procured and upgraded as described in action item 1, or forwards the information requested in action item 1.

   b. confirms that the CB and MC replacement actions requested in action item 2 have been completed, or provides an appropriate schedule for completion of these actions and confirms that a justification for continued operation has been completed and is being retained for possible NRC audit.

   c. confirms action item 3 has been completed.

2. All holders of a construction permit for nuclear power reactors are required to, within 120 days of receipt of this bulletin, provide a written report that:

a. confirms that no molded case CBs or MCs have been procured and upgraded as described in action item 1, or forwards the information requested in item 1.

b. confirms that the CB and MC replacement action requested in action item 2 will be completed prior to fuel load.

c. confirms action item 3 has been completed.

3. All addressees are requested to provide a report, within 180 days of receipt of this bulletin, that confirms that no molded case CBs or MCs have been procured for non-safety related applications as described in action item 4, or that describes the program required under action item 4.

The written reports required above shall be addressed to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended. In addition, a copy shall be submitted to the appropriate Regional Administrator.

This request is covered by Office of Management and Budget Clearance Number 3150-0011 which expires December 31, 1989. The estimated average burden hour is 400 man-hours per plant response, including assessment of the new requirements, searching data sources, gathering and analyzing the data, and preparing the required reports. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to the Office of Management and Budget, Room 3208, New Executive Office Building, Washington, D.C., 20503, and to the U.S. Nuclear Regulatory Commission, Records and Reports Management Branch, Office of Administration and Resource Management, Washington, D.C., 20555.

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Paul Gill, NRR (301) 492-0811
Jaime Guillen, NRR (301) 492-1170

Attachment 1: List of Recently Issued NRC Bulletins

UNDERWRITERS LABORATORIES INC.
333 PFINGSTEN ROAD · NORTHBROOK, ILLINOIS 60062–2096
an independent, not-for-profit organization testing for public safety

October 25, 1988

Mr. Paul Gill
Electrical Systems Branch
Mail Stop 8D20
Office of Nuclear Reactor Regulations
U.S. Nuclear Regulatory Commission
1 White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

Dear Mr. Gill:

Subject: NRC Draft Bulletin Dated October 12, 1988
Re: Rebuilt Circuit Breakers

Attached please find Underwriters Laboratories' comments on the above Subject Draft Bulletin from NRC.

Please feel free to call me if you have any questions, of if I can be of assistance to you in any way.

Sincerely,

S. Bhatia

S. Joe Bhatia
Vice President
Governmental Affairs

SJB:slr
cc: William Brach-NRC
Alex Marion-NUMARC
Russell Bell-NUMARC
Jack Bono-UL
Bob Baird-NEMA

UNDERWRITERS LABORATORIES INC.
333 PFINGSTEN ROAD · NORTHBROOK, ILLINOIS 60062-2096
an independent, not-for-profit organization testing for public safety

October 25, 1988

Electrical Division Branch (SELB)
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Paul Gill

Subject: Nuclear Regulatory Commission Draft Bulletin Dated October 12, 1988 - Molded-Case Circuit Breakers

Dear Mr. Gill:

The performance requirements for circuit breakers as outlined in the UL Standard for Safety, Molded-Case Circuit Breakers and Circuit Breaker Enclosures (UL489), and in UL's Follow-Up Service Programs include extensive testing at fault-current levels. Such fault-current level tests are necessarily destructive and a very important part of a full evaluation of a molded-case circuit breaker's ability to perform as intended.

The stated purpose of the bulletin UL was asked to review is to establish a test program for the evaluation of molded-case circuit breakers for which there is no verifiable traceability to the original circuit breaker manufacturer. The goal, as we understand it, is to have a test program which will provide some assurance that tested, nontraceable, circuit breakers are suitable for the intended service. Since all nontraceable circuit breakers which meet the performance tests are to be returned to service, another goal of the program is to conduct only those tests which would not severely degrade the condition of the circuit breakers tested.

Since the proposed program is to be nondestructive in nature, it is UL's opinion that the test program is not adequate to provide assurance that the tested, nontraceable,

Phone (312) 272-8800 MCI Mail No. 254-3343 Cable ULINC NORTHBROOK, IL MCI Telex No. 6502543343 FAX No. (312) 272-8129

circuit breakers would be suitable for the intended purpose. Therefore, UL's comments on the proposed tests, which you requested, should not be construed as an endorsement.

****

GENERAL TEST COMMENTS:

The methods of conducting the tests as outlined are not described. Since detailed procedures are needed to promote repeatability and insure usable test results, we recommend that the test methods be included or that reference be made to the test methods contained in the source standards.

The test program is identified as applicable to safety grade (Class 1E) circuit breakers. The test program should be conducted on all circuit breakers regardless of their application as the ability of a circuit breaker, commercial or safety grade, to sense and operate under expected overload conditions is important. Therefore, any noncompliance would show that the tested circuit breaker is not suitable for any application.

SPECIFIC TEST COMMENTS:

2.1 Mechanical Test

The described test can be used to determine that the circuit breaker latch and unlatch mechanism is mechanically functioning.

2.2 Time-Overcurrent Trip Test

This series of tests is intended to demonstrate the functioning of the circuit breaker in accordance with its time-current curve.

2.2.1 Rated Current Hold-In Test

At 100% rated current in a 40 degrees C ambient, the circuit breaker under test would not nuisance trip at maximum expected continuous currents.

The alternative test at 135% rated current in a 25 degrees C ambient would not be appropriate as a hold-in test. UL Listed circuit breakers are required to trip within the time period given under the test conditions stated.

If it is desired to establish another point on the time-current curve, a separate overcurrent trip test at 135% rated current in a 25 degrees C ambient would be appropriate with the maximum one-hour and two-hour trip times.

2.2.2 Overload Test

The test seems to combine two features of circuit breaker performance.

At 600% rated current in a 25 degrees C ambient, the test would check another point on the time-current curve. For such a test, closing on the 600% rated current would not be necessary.

A single operation load test at 600% rated current though informative would not, in our opinion, sufficiently demonstrate the normal load interrupting capabilities of the circuit breaker under test. The proposed test, if conducted at rated voltage and 600% of rated current, could cause degradation of the circuit breaker tested.

2.2.3 Time Delay Overcurrent Trip

This test is only suitable for evaluating additional points on the time-current curve of the circuit breaker under test.

2.3 Instantaneous Trip Test

This test is only suitable for evaluating additional points on the time-current curve of the circuit breaker under test.

2.4 Individual Pole Resistance Test

This test, if performed as defined in the NEMA AB-2 standard, could give an indication of the condition of the contacts and internal connections in each pole of the circuit breaker under test.

2.5 Dielectric Test

A dielectric voltage-withstand test could give an indication of the condition of the insulation system in the circuit breaker under test.

****

We hope these comments will be useful in addressing the expressed concerns over the operating capabilities of circuit breakers, not traceable to the original manufacturer which are presently both in use and in stock. However, it is our opinion that tests intended to be part of an inspection and maintenance program are not a substitute for a full test program which fully evaluates the interrupting capability of the involved circuit breakers.

Very truly yours,

S. G. Roll

S. G. ROLL
Vice President &
Chief Engineer
Electrical Division

SGR:PLR

GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose CA 95125

October 28, 1988
GBSLTR.044/PWM88-162
MFN88-69

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Subject: Comments on Draft NRC Bulletin on Molded Case Circuit Breakers, Dated 10/12/88

Attention: Brian K. Grimes, Director
Division of Reactor Inspection and Safeguards

Gentlemen:

GE Nuclear Energy has reviewed the draft Bulletin on Molded Case Circuit Breakers and is providing comments for your consideration prior to your issuance of a Bulletin on this subject.

We have four general comments for your consideration.

First, a more appropriate time frame for purchase records review by licensees would be three years (1986, 87 & 88). This time frame is consistent with the NRC guidance for retention of programatic nonpermanent records (Reg. Guide 1.28 Rev. 3), is sufficient for substantial licensee and vendor response from currently available commercial records, and is the same time frame used during recent NRC investigation activities. These records should provide a significant population for NRC assessment after utility review and feedback, including some testing results. Following evaluation of the results of an industry wide 3 yr. review, any expansion of this time frame, as already provided for in the draft bulletin, would have a sound basis.

Second, the adequacy of any proposed testing program should be endorsed by the industry organizations most knowledgeable about these circuit breakers, i.e., NEMA, UL, and the OEMs. We are aware that you are consulting with some of these organizations.

Third, the testing sequence should heed the recommendations contained in the NEMA letter of October 24, 1988 to William Brock, NRC.

Fourth, the testing program to verify the adequacy of circuit breakers already at the licensee facilities should not also be applied for dedication of future commercial purchases of circuit breakers known to have authentic OEM design, process and material control. Industry should be allowed to establish acceptable programs for dedication of such circuit breakers and these programs may rely on OEM design, process, and material controls. Such programs should be consistent with an overall equipment dedication program that is NRC audited. These programs should not be limited to a testing sequence as identified in the draft bulletin.

The attached markup addresses a few specific details in the draft bulletin.

Thank you for the opportunity to comment. If you would like to discuss this further, please call me at (408) 925-6948, or George Stramback at (408) 925-913.

P. W. Mar‑iott
Manager, Licensing & Consulting Services

Attachment

cc: R. [illegible]. Mitchell
A. [illegible]ion (NUMARC)
G. [illegible]. Stramback

Attachment

components have not been introduced and passed through the upgrading process. Further, the redundancy of safety systems and the in-service use of these components provide a reasonable basis for accepting installed replacement components that have been procured as safety grade from the OEM or from a corporate division associated with the OEM.

The NRC believes that the actions described in this bulletin need not extend to the electrical equipment (Class 1E and non-Class 1E) originally installed, at this time. This equipment appears to have been procured during plant construction from original manufacturers with full certification. The large quantities of electrical assemblies or components procured under bid packages minimizes the possibility of small vendors doing refurbishing business having supplied original equipment.

The actions requested in this bulletin apply only to molded case CBs procured within the last five years from the date of this bulletin for replacement for plant equipment, for plant modifications, or for maintaining as stored spares, as indicated in action item 1 below. However, the NRC investigation of this issue is not complete. A supplement to the bulletin may be issued to include other electrical equipment or a longer review period for procurement if warranted by the results of the ongoing evaluation or the results of testing called for by this bulletin.

Actions Requested

1. All addressees are requested to review their records of purchases for the last 5 years from the date of this bulletin to identify the number, types and applications of molded case CBs which have been installed as replacements for plant equipment, installed during modifications, or are being maintained as stored spares, that were:

3

10/12/88
DRAFT
BILB

equipment, installed during modifications, or are being maintained as stored spares, that were not procured from the original manufacturers or whose original source has not been determined, to assure that they are suitable for the intended service. A joint industry program which attains this objective is encouraged.

5. Information generated during the completion of items 1,2,3 and 4 above shall be documented and maintained for possible NRC audit.

6. Molded case CBs installed after the date of this Bulletin in safety related applications should be:

   a. manufactured by and procured from an OEM under a 10 CFR 50 Appendix B program; or

   b. procured from an OEM or others with verifiable traceability to the OEM, meeting applicable industry standards and upgraded to Class 1E by the licensee or others using appropriate tests ~~equivalent to those of Attachment 1. However, it should be noted that Attachment 1 does not address~~ In addition, seismic ~~or~~ and environmental qualification requirements ~~and that this~~ must be addressed, which would require additional testing or analysis based on plant-specific considerations to assure the CB is suitable for its intended application.

Reporting Requirements:

1. All holders of operating licenses are required by March 1, 1989 to provide a written report that:

   a. confirms that no molded case CBs have been procured or upgraded as described in action item 1, or forwards the information described in action item 1.

National Electrical Manufacturers Association
2101 L Street, N.W. Suite 300
Washington, D.C. 20037 (202) 457-8400
Telex 904077 NEMA WSH

October 24, 1988

Mr. William Brach
Chief of Vendor Inspection ~~Branch~~
Division of Reactor Inspection and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Draft Bulletin Dated October 12, 1988
Re: Rebuilt Circuit Breakers

Dear Mr. Brach:

NEMA would like to offer the following comments regarding the subject draft bulletin:

GENERAL

We believe that the term "circuit breaker manufacturer" should be substituted for "original equipment manufacturer" since the term "original equipment manufacturer" could also apply to a panelboard, switchboard or control equipment manufacturer who has purchased circuit breakers elsewhere.

DESCRIPTION OF CIRCUMSTANCES

NRC has limited the scope of this bulletin to only those breakers that have been purchased as safety-grade, or purchased as commercial product for upgrade to safety-grade, within the last five years.

We do not feel that, considering the possible failure modes, such limitations should be applied. Although, of course, priority should be given to safety-grade applications, any breaker that cannot be traced to the original circuit breaker manufacturer should be removed from service for inspection and test. The manufacturer of the breaker in question should be contacted for specific recommendations.

ACTIONS REQUESTED

1) Time Table

The bulletin indicates a specific time table, for records review, inspection, test and reporting.

Although we realize that such actions take time, we must reiterate our prior statement that tests of breakers collected from the companies in question seem to indicate that little calibration, testing, and inspection to ensure proper reassembly was done after the breakers were rebuilt. Many of the breakers would not meet the original specifications for calibration and performance and some experienced phase to phase faults at overload test levels. That would indicate that a serious hazard may exist where such breakers have been applied. Therefore, each rebuilt breaker should be removed from service for inspection and test. Priority should be given to safety related applications. Therefore, we urge that such actions be taken with all possible haste.

2) Addressees Requested to Provide Testing/Verification Program

Item 3 under "Actions Requested" requests licensees to undertake a testing program to assure the qualification of the equipment (or to assurance compilance with the appropriate requirements). Item 2 on page 7, under reporting requirements, requests holders of construction permits to demonstrate that rebuilt equipment is acceptable for such use.

Regarding Item 3 (and, to the extent that the demonstration of acceptability for intended use involves testing, Item 2):

An inspection and test program can give a reasonable degree of assurance of proper operation of a molded case circuit breaker only if the breaker has been manufactured under controlled conditions to conform to a proven design. This is the basis of manufacture of new circuit breakers.

A particular design of breaker is proven acceptable through inspections and type-tests. Included are both non-destructive and destructive tests (for performance characteristics such as endurance and short circuit capabilities).

Assurance that ongoing production breakers will continue to perform as the tested breakers did is obtained by controlling the manufacture of these breakers so they they are built the same as the tested breakers. Additional assurance is obtained through periodic inspection and test (both non-destructive and destructive) of representative products by both the manufacturer and by a third-party certifying authority such as Underwriters Laboratories.

There is no assurance that a rebuilt breaker has been rebuilt under controlled conditions to conform to a proven design.

On the contrary, each rebuilt breaker may be a unique combination of parts. Lubricants and sealants may be of the wrong type, wrong amount, improperly applied, or missing completely. Parts may be missing. Parts may be worn to near end of life. Parts may have been taken from breakers of several different designs or vintages.

Non-destructive tests and external inspections of a sealed breaker cannot detect such internal nonconformances which may adversely affect durability and short-circuit interrupting capability.

Because each rebuilt breaker can be unique, it is not possible to infer anything about the performance of a rebuilt breaker by destructively testing another breaker, whether that breaker is new or rebuilt.

To summarize our comments regarding the request that a test program be provided:

1. Determination of the critical performance characteristics of durability and short-circuit capabilities of circuit breakers requires destructive testing.

2. Such destructive testing must be performed on breakers known to be identical to the breaker to be placed in service.

3. The degree of similarity of rebuilt breakers to original design requirements, or of one rebuilt breaker to another, is unknown.

4. Durability and short circuit capabilities of rebuilt breakers are, therefore, unknown.

Since testing alone of rebuilt breakers does not assure that the breaker will meet all performance requirements, references in the Information Notice to a generic testing program to assure qualification, or to ensure compliance with the appropriate requirements, should be eliminated. The very most that could be expected from the testing program noted in the Information Notice, is a negative screening effect which would simply provide an indication that some of the breakers tested are definitely deficient. Such a program alone could not adequately assure that those breakers that passed the tests were acceptable for commercial service, much less for use in safety grade applications.

Further, and for the same reasons, evaluations by the NRC of the suitability of rebuilt circuit breakers should not be based on testing of the rebuilt breakers. Nor should they be based on similarity of the rebuilt breakers to original equipment breakers unless they can be shown to be identical in all respects.

3) Upgrades in Breaker Performance Must Be Considered

Users of rebuilt circuit breakers must also be aware that some circuit breaker interrupting ratings have been increased over the years. For example, a certain circuit breaker which currently has a 480 volt interrupting rating of 50,000 amperes was originally rated at 35,000 amperes. A rebuilt breaker, of that same type, acquired today might well be an older model with 35,000 amperes interrupting capacity, yet be installed on an application requiring 50,000 amperes interrupting capacity. The rebuilt breaker could meet all of its original design requirements, yet fail disastrously in service.

PROPOSED TEST PROGRAM

Once again, we do not believe that the suggested test program will assure adequate performance in all of the critical areas addressed by UL 489 and manufacturers test procedures. For example, intentional or inadvertent modification of the inside construction details of the circuit breaker could result in catastropic failure under short-circuit conditions but would not be detected by the abbreviated program suggested. Removal of the mechanism lubrication by cleaning solvents could result in early failure to operate, but again would not be detected by the proposed test program. Assurance of full performance through testing could only be established by a complete UL 489 test program, which would be destructive in nature.

The following comments address the suggested program as written and not its adequacy in assuring performance. The paragraph numbers refered to are those included in the proposed procedure.

Please note that special equipment, special laboratories, and trained personnel will be required to reliably implement even this abbreviated test program; particularly in regard to the proposed full-voltage overload test program.

CAUTION: Fault currents achieved during full-voltage overload tests can approach maximum available fault current levels. Such tests are normally conducted only in high-current laboratories by trained personnel utilizing special equipment and safety techniques.

2.0 - Test Procedures for Circuit Breakers

Tests must be conducted according to UL 489 with regard to procedure, wire size, set-up, equipment, etc. Experience has shown that test programs not closely following UL 489 end up using incorrect conductor sizes and test methods.

This paragraph states that tests should be performed in the order of the sequence listed. We suggest modifying the sequence to place the overload test 2.2.2, after completion of all calibration tests and before the dielectric test. The overload test, at full voltage, will require a special set-up in a high voltage laboratory. There is no point is incurring the expense in time and dollars if the breaker will not pass calibration tests. Also the rated current hold-in test, 2.2.1, with its long test times, should be placed after the time delay over-current trip, 2.2.3. The attached copy of the test program shows the suggested test sequence.

2.2.1 - Rated Current Hold-In Test

The first paragraph is in error in proposing a 135% rated current at 25^{0}C. It should be 110% rated current at 25^{0}C.

We recommend, however, that the test described in the first paragraph be run at 100% rated current at 25^{0}C as recommended in NEMA AB-2.

The second paragraph requires the test to be conducted for one-hour for circuit breakers rated 50 amperes or below and two hours for circuit breakers rated over 50 amperes. We recommend that the ampere rating be 100 amperes in both cases, in accordance with AB-2.

2.2.3 - Time Delay Over-Current Test

The final paragraph states that if a circuit exceeds the trip time shown in Table 1 it is unacceptable for Class 1E safety-grade applications. It is our opinion that a breaker exceeding the trip times in Table 1 is unacceptable for <u>any</u> application.

2.3.1 - Fixed Instantaneous Setting Breakers

2.3.2 - Adjustable Instantaneous Setting Circuit Breakers

In instantaneous testing of circuit breakers, we recommend that the maximum trip time be set at .167 seconds (10 cycles) in accordance with NEMA AB 2.

The tolerance on instantaneous trip setting as a percent of that shown on the manufacturer's trip time curves should apply only to factory or laboratory tested circuit breakers, again per AB 2. If breakers are to be tested in the field, the broader tolerances of 140% and 70%, as listed in NEMA AB-1. should be used. This broader tolerance is necessa.y because of less controlled test conditions and equipment typically used for field testing. In any case, instantaneous tests should be run by the pulse method, and not the so called run-up method. While the run-up method utilizes simpler, less expensive, equipment, it relies heavily on the skill and experience of the operator and can produce erratic results.

2.4 - Individual Pole Resistance Test

Experience has shown that measurement of the very small contact resistance of each pole is unreliable and depends heavily on the precision of the instruments used. A better approach is to check and compare millivolt drop readings with data provided by the manufacturer. We known of no basis for disqualifying breakers where the differences in readings from pole to pole exceed 50%.

We strongly suggest that the pole resistance test be replaced by a connector temperature test. When the rated current hold-in test (2.2.1) is being run, a temperature probe can be used to measure line and load connector temperatures. They should not exceed a 50^{0}C rise over the ambient. Such tests would give a much better indicator of potential overheating than either contact resistance or millivolt data.

2.5 - Dielectric Tests

We know of no basis for the use of 2,500 volts D.C. as a dielectric test. We recommend the A.C. test since similar or equivalent tests are run by the manufacturers and because test equipment is much more readily available.

SUMMARY OF COMMENTS

1) Tests have indicated that serious hazard may exist where rebuilt circuit breakers have been applied.

2) We believe that any breaker that can not be traced back to the original circuit breaker manufacturer should be removed from service for inspection and test regardless of age or application. The original circuit breaker manufacturer should be contacted for specific recommendations.

3) Performance of rebuilt breakers can not be assured through use of a generic non-destructive test program. Only destructive testing would provide the necessary assurances. The proposed test program would provide only a first-cut negative screening.

4) Even as a first-cut negative screen, the proposed test program has some technical errors which should be addressed. Further, the proposed test program involves tests which require special personnel, equipment, and safety techniques.

We appreciate the opportunity to comment on the draft bulletin and stand ready to work with NRC in bringing this issue to a speedy resolution.

If you have any questions regarding our comments, please feel free to give me a call.

Sincerely,

Robert W. Baird

Robert W. Baird
Division Staff Executive

Attachment

cc: J. Bhatia, UL
P. Gill, NRC
A. Marion, NUMARC
T. F. on Rebuilt Breakers
5-AB Section

PROPOSED SEQUENCE

2.0 Test Procedures for CBs

The following tests should be performed in the order of sequence listed.

2.1 Mechanical Test

The CB should be ~~tripped~~ operated, reset and closed a minimum of five times to insure that the latching surfaces are free of any binding.

2.2 Time-Overcurrent Trip Test

2 ~~2.2.1~~ Rated Current Hold-In Test

This test should be conducted at 100% rated current and at an ambient air temperature of 40°c ± 3°c, or at 125% rated current and at an ambient temperature of 25°c ± 3°c. 110%

Equal current should be applied to all poles of the CB; and CB must not trip within 1 hour for CBs rated 50 amperes or below and within 2 hours for CBs rated over 50 amperes. 100 100

5 ~~2.2.2~~ Overload test

This test consists of 1 operating cycle (i.e., closing action followed by an opening action) of the CB at 600% rated current. There shall be no electrical or mechanical breakdown of the CB during this test.

2.1 ~~2.2.3~~ Time Delay Overcurrent Trip

This test should be conducted at an ambient air temperature of 25 degrees c ± 3°degrees c.

Enclosure

A current of 300 percent of the marked rating should be applied to each pole of the CB. The trip time for each pole should be compared against the time shown in the CB manufacturer's time-current curves. If the test trip times obtained for each pole are not within the time band shown on the CB manufacturer's time-current curves, then the test trip must not exceed the time specified in table 1 and the acceptance of CB evaluated with the criteria as listed below:

TABLE 1
VALUES FOR OVERCURRENT TRIP TEST
(AT 300% OF RATED CONTINUOUS CURRENT OF CIRCUIT BREAKER)
(REF. NEMA AB-2)

Breaker Voltage Volts	Range of Rated Continuous Current Amperes	Maximum Trip Time In Seconds
240	15-45	50
240	50-100	70
600	15-45	70
600	50-100	125
240	110-225	200
240	250-400	300
600	110-225	250
600	250-400	300
600	450-600	350
600	700-1200	500
600	1400-2500	600
600	3000-5000	650

Minimum trip times: If the minimum tripping times are lower than indicated by the manufacturer's time-current curves for the breaker under test, the breaker should be retested after it has been cooled to 25°C. If the values obtained are still lower after retest, the coordination with upstream and downstream breakers should be evaluated. If no miscoordination is indicated, then the CB is acceptable.

Enclosure

Maximum tripping time: If the tripping time exceeds the maximum tripping time shown on the manufacturer's time-current curves but is below the time shown in table 1 above, check the breaker time against the protection requirements of the circuit (such as cable, penetration, etc.) to ensure that the CB provides the protection, as well as the coordination with upstream and downstream CBs. If the CB provides the necessary protection and coordination, then the CB is acceptable.

Maximum allowable time: If the tripping time of the CB exceeds the trip time shown in the table 1 above, the breaker is unacceptable ~~for class 1E applications~~.

2.3 Instantaneous Trip Test

2.3.1 Fixed Instantaneous Setting CBs

140 %

70 %

Each pole of the CB should be tested for pickup of the instantaneous unit. The average of the three readings for each pole must be between 80% and 120% of the instantaneous trip setting as shown on the manufacturer time-current curves. The trip time should not exceed .05 seconds (3 cycles).

2.3.2 Adjustable Instantaneous Setting CBs

Same as 2.3.1 except that each pole must be tested at the lowest and highest setting.

140

70

The average trip value for the lowest setting should be between 75 percent and 125 percent, and the highest setting should be between 80 percent and 120 percent of the setting value shown on the manufacturer's time-current curves.

75

140

[illegible]

2.3.3 Short-time trip setting test

This test is applicable only if the CB is equipped with the short-time delay trip. This test should be conducted at an ambient air temperature of 25°c ± 3°c. The operation of the short-time delay unit should be within 90% and 125% of the overcurrent setting of the CB as shown on the manufacturer's time-current curves.

2.4 Individual Pole Resistance Test

The contact resistance of each pole of the CB should be measured at ambient temperature. The average of 3 readings for each pole should be calculated and compared with the manufacturer's data or with those of similar CBs of the same manufacturer. Also, the average reading of each pole should be compared with each other and the difference between the pole readings should not exceed fifty percent.

2.6 ~~2.5~~ Dielectric Tests

The dielectric test should be conducted at an ac test voltage of 1760 volts (0.8 x [2 x Rated Voltage + 1000 volts]) ~~, or 2500 volts~~ for 1 minute withstand. The dielectric test should be conducted for (1) line to load terminals with CB open, (2) line to line terminals with CB closed, and (3) pole to ground with CB open and closed.

OMB No.: 3150-0011
NRCB 88-10

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

November 22, 1988

NRC Bulletin No. 88-10: NONCONFORMING MOLDED-CASE CIRCUIT BREAKERS

Addressees:

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose:

The purpose of this bulletin is to request that addressees take actions to provide reasonable assurance that molded-case circuit breakers (CBs), including CBs used with motor controllers, purchased for use in safety-related applications without verifiable traceability[1] to the circuit breaker manufacturer (CBM)[1] perform their safety functions.

Description of Circumstances:

NRC Information Notice No. 88-46, "Licensee Report of Defective Refurbished Circuit Breakers," dated July 8, 1988 and Supplement 1 thereto, dated July 21, 1988, discussed a report by Pacific Gas and Electric Company that indicated that its Diablo Canyon Nuclear Power Plant was supplied 30 CBs by Anti-Theft Systems, Inc. through a local electrical distributor. These CBs (Square D molded-case, type KHL 36125) were intended for use in non-safety-related applications at the Diablo Canyon Nuclear Power Plant. Square D Company reported that an inspection and testing of these breakers determined that the CBs were refurbished Square D Company equipment. Furthermore, Square D reported that several of the circuit breakers tested did not comply with Square D or Underwriters Laboratories, Inc. (UL) specifications for all of the electrical tests performed. Information Notice No. 88-46 also listed several California companies that were involved in supplying surplus refurbished and possibly defective refurbished electrical equipment to the nuclear industry.

During recent NRC inspections, additional examples were identified that indicate a potential safety concern regarding electrical equipment supplied to nuclear power plants. The NRC is concerned that equipment being procured as new and assumed to meet all applicable plant design requirements and/or original manufacturer's specifications may, in fact, not conform to these requirements and specifications.

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1. Refer to Attachment 2 for Definition of Terms

The actions requested in this bulletin are limited to molded-case CBs. Molded-case CBs are tested and calibrated at the manufacturer's plant in accordance with recognized industry standards, such as UL-489, "Molded Case Circuit Breakers and Circuit Breaker Enclosures," and National Electrical Manufacturers Association (NEMA)-AB1, "Molded-Case Circuit Breakers." Since molded-case CBs have factory-calibrated and sealed elements, any unauthorized modification or refurbishing of these CBs could jeopardize their design capability and reliability.

The NRC is concerned that the reliability and capabilities of refurbished CBs purchased as commercial grade (non-Class 1E) for later upgrading to safety-related (Class 1E) applications may not meet the minimum commercial grade standards. In addition, the NRC is concerned about the reliability and capabilities of commercial grade CBs upgraded to safety-related because of some observed inadequacies in the dedication process and numerous failures found during the testing of some of these CBs. In order to properly dedicate electrical items procured as commercial grade for subsequent use in safety-related applications, the dedication process should build from the commercial grade quality, include a proper evaluation of seismic and environmental qualification, confirm critical parameters, and include testing as appropriate.

Molded-case CBs purchased from the CBM or that can be traced to the CBM are of lesser concern than other molded-case CBs because CBs from the CBM, whether safety-related or commercial grade, are manufactured under controlled conditions to conform to a proven design. Safety-related CBs purchased as Class 1E from the CBM are controlled under quality assurance (QA) programs that conform to Appendix B of 10 CFR Part 50. While upgrading programs may vary in quality, the controls exercised by the CBM over the manufacturing activities provide reasonable assurance that improperly refurbished components have not been introduced and passed through the upgrading process. Furthermore, the redundancy of safety systems and the in-service use of CBs provide a reasonable basis for accepting installed replacement CBs that have been procured from the CBM or that can be traced to the CBM.

The NRC currently believes that the concerns addressed in this bulletin do not apply to electrical equipment (safety-related and commercial grade) originally installed in plants. This equipment appears to have been procured during plant construction from CBMs with full certification. The large quantities of electrical assemblies or components procured under bid packages during plant construction reduce the possibility of any original plant equipment being supplied by vendors doing refurbishing.

Although the actions requested in this bulletin only apply to safety-related molded-case CBs, the NRC intends to monitor industry programs to ensure that other molded-case CBs, which may have been installed as replacements, installed during modifications, or are being maintained as stored spares, are suitable for their intended service. Addressees are encouraged to participate in a joint program. If industry programs are either not timely or not sufficient, additional regulatory actions will be taken, as appropriate.

The NRC requested and received comments from the Nuclear Management and Resources Council (NUMARC), the National Electrical Manufacturers Association (NEMA), and the Underwriters Laboratories, Inc. (UL), during the preparation of this bulletin. These comments were considered and some were appropriately incorporated into this bulletin.

NEMA has commented to the NRC that determination of the critical performance characteristics of durability and short-circuit capabilities of CBs requires destructive testing of selected breakers that are representative of CBs to be placed in service. Because a refurbished breaker may not have been refurbished under controlled conditions to conform to a proven design, destructively testing selected breakers will not infer anything about a refurbished CB. UL provided specific comments on the tests in Attachment 1 of this bulletin. In addition, they stated that, "it is UL's opinion that the test program is not adequate to provide assurance that the tested, non-traceable, circuit breakers would be suitable for their intended purpose." Although the test program described in Attachment 1 of this bulletin does not provide complete verification of all the performance requirements and characteristics of molded-case CBs (such as seismicity or fault clearing capability), the NRC considers the test program to provide a reasonable assurance of performance requirements and characteristics most important to ensuring reactor safety. This, considered in conjunction with (1) the limited number of nonconforming CBs that may remain installed in safety-related systems following implementation of the actions requested by this bulletin, (2) the existence of redundant safety-related systems in nuclear power reactors that are required by NRC regulations, (3) the license required in-service testing of installed CBs performed to demonstrate the CB's functional performance, and (4) the low frequency of occurrence of seismic events and severe electrical faults, provides a reasonable assurance that nuclear power reactors can be operated without undue risk to the health and safety of the public.

The NRC investigation of this issue is not complete. A supplement to this bulletin may be issued to include other electrical equipment or a longer procurement review period if warranted by the results of the ongoing evaluations or the results of testing requested in this bulletin.

Actions Requested:

1. All addressees are requested to perform the following review by March 1, 1989:

   a. Identify all molded-case CBs purchased prior to August 1, 1988, that are being maintained as stored spares for safety-related (Class 1E) applications or commercial grade CBs that are being maintained as stored spares for future use in safety-related applications; this includes CBs purchased from a CBM or from any other source. If the number of these stored spare CBs is less than 50 at a nuclear plant site, then randomly select CBs purchased between August 1, 1983 and August 1, 1988 that have been installed in safety-related applications as replacements or modifications to form a minimum sample of 50 CBs per nuclear plant site.

b. Verify the traceability of these CBs.

c. Identify the number, manufacturer, model number, and to the extent possible the procurement chain for all those CBs identified in (1a) that cannot be traced to the CBM. For installed CBs, also identify each system in which they are/were installed.

2. All holders of operating licenses who identify installed CBs per item 1 above or item 4 below that cannot be traced to a CBM are requested to prepare, within 30 days of the completion of each item, an analysis justifying continued operation until items 1 through 5 of the actions requested in this bulletin have been completed.

3. All addressees who identify 80 percent or more CBs traceable to the CBM per item 1 above are requested to test the CBs that are not traceable to the CBM in accordance with the test program described in Attachment 1. Any installed CBs that fail any of these tests should be replaced with CBs that meet the criteria of item 7 of the actions requested or CBs that pass all tests in accordance with the testing program described in Attachment 1. If more than 10 percent of the CBs tested fail any of the tests described in Attachment 1, continue with item 4; otherwise, proceed to item 6 of the actions requested.

   Holders of operating licenses are requested to complete this testing program before startup from the first refueling outage beginning after March 1, 1989. Holders of construction permits are requested to complete this testing program before fuel load.

4. All addressees who identify less than 80 percent of the CBs traceable to the CBM per item 1 above or who identify a failure rate of more than 10 percent for the CBs tested per item 3 above are requested to perform the following actions:

   a. Identify all molded-case CBs that have been purchased between August 1, 1983 and August 1, 1988, and installed in safety-related applications as replacements or installed during modifications.

   b. Verify the traceability of these CBs.

   c. Identify the number, manufacturer, model number, system in which they are/were installed, and to the extent possible, the procurement chain for all those CBs identified in (4a) that cannot be traced to the CBM.

5. All addressees who identify installed CBs that cannot be traced to the CBM per item 4 above are requested to replace these CBs with components that meet the criteria of item 7 of the actions requested or to test them in accordance with the program described in Attachment 1; CBs that fail any of these tests should be replaced with CBs that meet the criteria of item 7 of the actions requested or CBs that pass all tests in accordance with the test program described in Attachment 1.

Holders of operating licenses are requested to replace or to test at least one-half, or all if the total number is less than 75, of these installed CBs before startup from the first refueling outage beginning after March 1, 1989. The remaining CBs should be replaced or tested before startup from the second refueling outage beginning after March 1, 1989.

Holders of construction permits are requested to replace or to test these installed CBs before fuel load.

6. Information generated while performing the actions requested in items 1, 2, 3, 4, and 5 above should be documented and maintained for a period of 5 years after the completion of all requested actions.

7. With the exception of actions taken in response to items 3 and 5 of the actions requested above, molded-case CBs installed in safety-related applications after August 1, 1988 should be:

   a. Manufactured by and procured from a CBM under a 10 CFR 50, Appendix B, program; or

   b. Procured from a CBM or others with verifiable traceability to the CBM, in compliance with applicable industry standards, and upgraded to safety-related by the licensee or others using an acceptable dedication program. The NRC encourages addressees to significantly upgrade their dedication programs through a joint industry effort to ensure their adequacy and consistency. The NRC will monitor these industry initiatives and if they are not sufficient or not timely, or if problems with the dedication of commercial grade equipment for safety-related use continue, the NRC will take appropriate regulatory actions.

8. Addressees that cannot meet the schedule for the actions requested above and/or the corresponding reporting requirements below, should justify to the NRC their proposed alternative schedule.

Reporting Requirements:

1. All holders of operating licenses are required to provide a written report by April 1, 1989, that:

   a. Confirms that only molded-case CBs that meet the criteria of item 7 of the actions requested are being maintained as stored spares for future use in safety-related applications.

   b. Summarizes the total number, manufacturer, model number, and to the extent possible the procurement chain of those CBs that could not be traced to the CBM in items 1 and 4 of the actions requested. For installed CBs, also identify each system in which they are/were installed. If item 4 of the actions requested has not been completed by April 1, 1989, due to the schedule for tests in item 3 of the actions requested, this information should be updated within 30 days of the completion of item 4 to address those additional CBs that could not be traced to the CBM.

c. Confirms that items 1, 2, 3, 4, 5, 6 and 7 of the actions requested have been completed or will be implemented as requested.

2. All holders of operating licenses are required to submit a report that summarizes available results of tests conducted in accordance with items 3 and 5 of the actions requested within 30 days after startup from the first and second refueling outages beginning after March 1, 1989. For CBs that pass these tests, the only information required is the number, manufacturer, model number, and to the extent possible the procurement chain of CBs tested (summary report format is acceptable). For CBs that fail these test(s), these reports should indicate the test(s) and the values of test parameter(s) at which the failure(s) occurred, as well as the corresponding manufacturer, model number, and to the extent possible, the procurement chain.

3. All holders of construction permits are required to provide a written report by April 1, 1989, that:

   a. Confirms that only molded-case CBs that meet the criteria of item 7 of the actions requested are being maintained as stored spares for future use in safety-related applications.

   b. Summarizes the total number, manufacturer, model number, and to the extent possible the procurement chain of those CBs that could not be traced to the CBM in items 1 and 4 of the actions requested. For installed CBs, also identify each system in which they are/were installed. If item 4 of the actions requested has not been completed by April 1, 1989, due to the schedule for tests in item 3 of the actions requested, this information should be updated within 30 days of the completion of item 4 to address those additional CBs that could not be traced to the CBM.

   c. Confirms that items 1, 3, 4, 5, 6 and 7 have been completed or will be implemented before fuel load.

4. All holders of construction permits are required to submit a report that summarizes the results of tests conducted in accordance with items 3 and 5 of the actions requested within 30 days after fuel load. For CBs that pass these tests, the only information required is the number, manufacturer, model number, and to the extent possible, the procurement chain (summary report format is acceptable). For CBs that fail these test(s), the report should indicate the test(s) and the values of test parameter(s) at which the failure(s) occurred, as well as the corresponding manufacturer, model number, and to the extent possible, the procurement chain.

The written reports required above shall be addressed to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended. In addition, a copy shall be submitted to the appropriate Regional Administrator.

This request is covered by Office of Management and Budget Clearance Number 3150-0011 which expires December 31, 1989. The estimated burden hour is 1000 to 10,000 man-hours per plant response, including assessment of these requirements, searching data sources, testing, and analyzing the data, and preparing the required reports. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to the Office of Management and Budget, Room 3208, New Executive Office Building, Washington, D.C., 20503, and to the U.S. Nuclear Regulatory Commission, Records and Reports Management Branch, Office of Administration and Resource Management, Washington, D.C., 20555.

If you have any questions regarding this matter, please contact one of the technical contacts listed below or the Regional Administrator of the appropriate NRC regional office.

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Attachments:
1. Test Program for Molded Case Circuit Breakers
2. Definition of Terms
3. List of Recently Issued NRC Bulletins

TEST PROGRAM FOR MOLDED-CASE CIRCUIT BREAKERS

1.0 Test Program Objectives

The objective of this proposed test program is to verify the reliability and capabilities of molded-case circuit breakers (CBs).

For the safety of personnel and others involved with the activities related to these proposed tests, appropriate safety practices, such as ANSI/NFPA 70E, "Electrical Safety Requirements for Employee Workplaces," Part II, should be followed.

These proposed tests have been based on tests described in industry standards, such as NEMA AB-1, "Molded-Case Circuit Breakers," NEMA AB-2, "Procedures for Field Inspection and Performance Verification of Molded-Case Circuit Breakers Used in Commercial and Industrial Applications," UL 489 "Molded Case Circuit Breakers and Circuit Breaker Enclosures," and NETA STD ATS-1987, "National Electrical Testing Association, Acceptance Testing Specifications."

2.0 Test Procedures for CBs

The following tests should be performed in the sequence listed. CBs failing any of these tests should be considered unacceptable for safety-related applications.

2.1 Mechanical Test

The CB should be operated, reset, and closed a minimum of five times, to ensure that the latching surfaces are free of any binding.

2.2 Individual Pole Resistance or Millivolt Drop Test (Ref. NETA STD ATS-1987 & NEMA AB-2)

The contact resistance of each pole of the CB should be measured at ambient temperature. Three readings of each pole should be taken with the CB operated without load between each reading. The average of three readings for each pole should be calculated and compared with the manufacturer's contact resistance data or with those values of similar CBs from the same manufacturer. Also, the average value for each pole should be compared with the average of the other poles and the difference between the pole values should not exceed 50 percent of the lowest value; or

A millivolt drop test may be performed by applying a direct current across the closed CB contacts and measuring the voltage drop due to the contact resistance. The millivolt drop test should be performed at room temperature. Direct current should be applied across each pole and the millivolt drop and test current recorded for each pole. Three readings of each pole should be taken with the CB operated without load between each reading. The average of the three readings for each pole should be calculated and compared with the manufacturer's value for acceptance of the breaker.

2.3 Rated Current Hold-In Test (Ref. NEMA AB-1 & UL 489)

This test should be conducted at 100% rated current and at an ambient air temperature of 25°C ± 3°C, and followed by a test at 135% rated current and at an ambient temperature of 25°C ± 3°C.

Equal 100% rated currents should be applied to all poles of the CB. The CB must not trip within 1 hour for CBs rated 50 amperes or below or within 2 hours for CBs rated over 50 amperes during this test. At the end of the 100% rated current test, the current should be increased to 135% and the CB should trip within 1 hour for CBs rated 50 amperes or below or within 2 hours for CBs rated over 50 amperes.

2.4 Overload Test (Ref. NEMA AB-1 & UL 489)

This test consists of one operating cycle (i.e., closing action followed by an opening action) of the CB at 600% rated current. This test may be conducted at low voltage. There should be no electrical or mechanical breakdown of the CB during this test.

2.5 Instantaneous Trip Test (Ref. NEMA AB-1 & UL 489)

2.5.1 Fixed Instantaneous Setting CBs

Each pole of the CB should be tested for pickup of the instantaneous unit. Each pole must be between 75% and 125% of the instantaneous trip rating. The trip time should not exceed 0.1 seconds (6 cycles).

2.5.2 Adjustable Instantaneous Setting CBs

This test is the same as that in Section 2.5.1 except that each pole must be tested at the lowest and highest settings.

The trip value for the lowest setting should be between 75% and 125% of the lowest setting, and the highest setting should be between 80% and 120% of the highest setting.

2.5.3 Short-Time Trip Setting Test

This test is applicable only if the CB is equipped with the short-time delay trip. This test should be conducted at an ambient air temperature of 25°C ± 3°C. The operation of the short-time delay unit should be within 90% and 125% of the overcurrent setting of the CB as shown on the manufacturer's time-current curves.

2.6 Time Delay Overcurrent Trip (Ref. NEMA AB-2)

This test should be conducted at an ambient air temperature of 25°C ± 3°C.

A current of 300% (at low voltage) of the marked rating should be applied to each pole of the CB. The trip time for each pole should be compared with the time shown in the CB manufacturer's time-current curves. If the test trip times obtained for each pole are not within the time band shown on the CB manufacturer's time-current curves, then the test trip must not exceed the time specified in Table 1 and the acceptance of the CBs must be evaluated with the criteria listed below:

TABLE 1
VALUES FOR OVERCURRENT TRIP TEST
(AT 300% OF RATED CONTINUOUS CURRENT OF CIRCUIT BREAKER)
(REF. NEMA AB-2)

Breaker Voltage Volts	Range of Rated Continuous Current Amperes	Maximum Tripping Time In Seconds
240	15-45	50
240	50-100	70
600	15-45	70
600	50-100	125
240	110-225	200
240	250-400	300
600	110-225	250
600	250-400	300
600	450-600	350
600	700-1200	500
600	1400-2500	600
600	3000-5000	650

Minimum Tripping Time: If the minimum tripping times are lower than indicated by the manufacturer's time-current curves for the CB under test, the CB should be retested after it has been cooled to 25°C. If the values obtained are still lower after retest, the coordination with upstream and downstream CB should be evaluated. If no problem with coordination is indicated, then the CB is acceptable.

Maximum Tripping Time: If the tripping time exceeds the maximum tripping time shown on the manufacturer's time-current curves but is below the time shown in Table 1, check the CB time against the protection requirements of the circuit (such as cable, penetration, etc.) to ensure that the CB provides the protection, as well as the coordination with upstream and downstream CBs. If the CB provides the necessary protection and coordination, then the CB is acceptable.

Maximum Allowable Time: If the tripping time of the CB exceeds the trip time shown in Table 1, the breaker is unacceptable for Class 1E applications.

2.7 Dielectric Tests (Ref. NEMA AB-1 & UL-489)

The dielectric test should be conducted at an ac test voltage of 1760 volts (80% x [2 x rated voltage + 1000 volts]), or at 2500 volts dc for 1 minute withstand. The dielectric test should be conducted for (1) line to load terminals with CB open, (2) line to line terminals with CB closed, and (3) pole to ground with CB open, and (4) pole to ground with CB closed.

Attachment 2
NRCB 88-10
November 22, 1988

DEFINITION OF TERMS

CIRCUIT BREAKER MANUFACTURER (CBM)

The manufacturing facility that actually produced the circuit breaker being purchased.

VERIFIABLE TRACEABILITY

Documented evidence such as a certificate of compliance that establishes traceability of purchased equipment to the CBM. If the certificate of compliance is provided by any party other than the CBM, the validity of such certificate must be verified by the licensee or permit holder through an audit or other appropriate means.

DEDICATION PROCESS

The process by which commercial grade (non-Class 1E) equipment is upgraded to safety-related (Class 1E) and is thereby considered qualified for use in safety-related applications. The dedication process must include:

a. A technical evaluation to determine the characteristics critical to fulfilling the safety function(s).

b. An acceptance process to ensure that those critical characteristics are met.

Attachment 3
NRCB 88-10
November 22, 1988

LIST OF RECENTLY ISSUED
NRC BULLETINS

Bulletin No.	Subject	Date of Issuance	Issued to
88-05, Supplement 2	Nonconforming Materials Supplied by Piping Supplies, Inc. at Folsom, New Jersey and West Jersey Manufacturing Company at Williamstown, New Jersey	8/3/88	All holders of OLs or CPs for nuclear power reactors.
88-08, Supplement 2	Thermal Stresses in Piping Connected to Reactor Coolant Systems	8/4/88	All holders of OLs or CPs for light-water-cooled nuclear power reactors.
88-09	Thimble Tube Thinning in Westinghouse Reactors	7/26/88	All holders of OLs or CPs for W-designed nuclear power reactors that utilize bottom mounted instrumentation.
88-08, Supplement 1	Thermal Stresses in Piping Connected to Reactor Coolant Systems	6/24/88	All holders of OLs or CPs for light-water-cooled nuclear power reactors.
88-08	Thermal Stresses in Piping Connected to Reactor Coolant Systems	6/22/88	All holders of OLs or CPs for light-water-cooled nuclear power reactors.
88-05, Supplement 1	Nonconforming Materials Supplied by Piping Supplies, Inc. at Folsom, New Jersey and West Jersey Manufacturing Company at Williamstown, New Jersey	6/15/88	All holders of OLs or CPs for nuclear power reactors.
88-07	Power Oscillations in Boiling Water Reactors (BWRs)	6/15/88	All holders of OLs or CPs for BWRs.

OL = Operating License
CP = Construction Permit