Mrs. Becke

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PROPOSED RULE
General Design Criteria

ATOMIC INDUSTRIAL FORUM INC.

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Secretary
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Sir:

Pursuant to notice which appeared in the Federal Register of July 11, 1967, the Forum Committee on Reactor Safety is pleased to forward the enclosed comments on AEC's proposed "General Design Criteria for Nuclear Power Plant Construction Permits".

These comments, which in a number of instances take the form of a redraft of the proposed criteria, are based on information developed during an August 9 meeting of the Committee. They have been further refined by a Committee task force comprised of the following members: Wallace Behnke of Commonwealth Edison Company; Arthur C. Gehr of Isham, Lincoln & Beale; R. J. McWhorter of General Electric Company; J. E. Tribble of Yankee Atomic Electric Company; Robert A. Wiesemann of Westinghouse Electric Corporation; and Edwin A. Wiggin of the Forum staff.

The comments have subsequently been circulated to those additional members of the Committee who participated in the August 9 meeting. It may, therefore, be concluded that the enclosed comments generally represent the views of the following additional Committee members:

R. H. Bielecki, Pennsylvania Power & Light Company
Warren S. Brown, Dilworth, Secord, Meagher & Associates, Ltd.
Harvey F. Brush, Bechtel Corporation
Robert W. Davies, Baltimore Gas and Electric Company
William S. Farmer, Allis-Chalmers Manufacturing Company
George C. Freeman, Jr., Hunton, Williams, Gay, Powell & Gibson
Robert E. Kettner, Consumers Power Company
R. W. Kupp, S. M. Stoller Associates
C. A. Larson, Consolidated Edison Company of New York, Inc.
Zelvin Levine, Hittman Associates, Inc.
James V. Neely, Jersey Central Power and Light Company
H. C. Ott, Ebasco Services, Inc.
Joseph W. Ray, Battelle Memorial Institute
Glenn A. Reed, Wisconsin Electric Power Company
Marlin Remley, Atomics International, Inc.
Royce J. Rickert, Combustion Engineering, Inc.

W. N. Thomas, Virginia Electric and Power Company
Robert E. Wascher, The Babcock & Wilcox Company
Samuel Zwickler, Burns & Roe, Inc.

Although these comments have been throughly reviewed by those individuals listed above, it should be understood that they do not necessarily represent a unanimity of opinion on all the criteria. Members of the Committee who participated in the August 9 discussion, particularly those who find themselves at variance with the views expressed herein, have been urged to make their views known directly to the AEC in behalf of their own respective companies and organizations.

Perhaps a further note of explanation on the enclosed comments is in order.

In the Committee's opinion, the proposed criteria are appreciably better organized than those initially suggested in November 1965. We have also noted with appreciation that some of the Committee's suggestions on the earlier criteria have been accommodated in the criteria now proposed.

The Committee believes that the principal objectives of the criteria should be to assist in the design of nuclear power plants, the preparation of applications for construction permits and operating licenses therefor and regulatory review of these applications to determine if such plants can be constructed and operated without undue risk to the health and safety of the public. The Committee further believes that these objectives should be explicitly stated and that they can be most effectively attained by writing the criteria to the extent possible as performance specifications.

We recommend that the following paragraph be added to the introduction - possibly following the last paragraph of the introduction as it appeared in the Federal Register notice:

> "Each of the requirements stated and implied in the criteria is premised on assuring that the nuclear power plant will be designed, constructed and operated in such a manner as not to cause undue risk to the health and safety of the public from radiation or the release of radioactive materials. To facilitate compliance with the requirements contained in the criteria, the criteria are presented to the extent possible, as performance specifications."

The Committee further believes that the introduction to the criteria should make more explicit reference to their intended direct applicability to water reactors in contrast to their only indirect applicability to reactors of other types, including fast breeders.

Some members of the Committee have noted the desirability and advantages of publishing these criteria as a guide rather than as an appendix to 10 CFR 50. They point out that, as a guide, their interpretation, application and refinement could be more easily adapted to a rapidly

ATOMIC INDUSTRIAL FORUM INC.

If questions arise in reviewing these comments, the members of the task force would be pleased to meet with representatives of the AEC regulatory staff.

Sincerely,

Edwin A. Wiggin

Edwin A. Wiggin
Committee Secretary

EAW:epb
Enclosure

Comments of Forum Committee on Reactor Safety on AEC's Proposed Construction Permit Criteria

CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required.

> In the first sentence we have modified "accidents" with "nuclear" and substituted the phrase "cause undue risk to the health and safety of the public" to more precisely reflect what we believe was the AEC's intent. In the last sentence of the original draft, we have dropped the word "sufficiency" since we do not believe that it should be the responsibility of the applicant to document this unless the sufficiency of some specific item is in question. If for any reason the AEC questions the adequacy or sufficiency of a code or standard, it should take this matter up with the appropriate code drafting committee. Note that we have added a sentence requiring a showing of adequacy where there is no applicable code. The balance of the suggested changes are editorial in nature.

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

> The changes in the first sentence are in line with those suggested for Criterion 1. We have deleted the word "additional" on the premise that it is not reasonable to ask the applicant to consider the simultaneous or cumulative forces of more than one extraordinary natural phenomenon.

<u>CRITERION 3 - FIRE PROTECTION (Category A)</u>

A reactor facility shall be designed such that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

> These changes are consistent with the objective of assuring that there will be no undue risk to the health and safety of the public.

<u>CRITERION 4 - SHARING OF SYSTEMS (Category A)</u>

Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public.

As originally drafted, this criterion made unacceptable any impairment of safety, whether the impairment was significant or insignificant. This is unreasonable. Some impairment will undoubtedly result from almost any sharing but the impairment may not be significant enough to preclude the sharing. The test should be whether the sharing will result in undue risk to the health and safety of the public.

CRITERION 5 - RECORDS REQUIREMENTS (Category A)

The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public.

Some of the records that should be maintained may or may not be under the physical control of the licensee or operator. He can, however, assure that they are maintained, by contractual arrangements, if necessary. Those records which are important are those which could have some bearing on the health and safety of the public.

CRITERION 6 - REACTOR CORE DESIGN (Categories A & B)

The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.

We assume that "acceptable fuel damage limits" will be based on "undue risk to the health and safety of the public", not on economic grounds. The latter consideration is a matter for the licensee to decide. Further, these limits will depend on the circumstances leading to the damage. The example "transient situations" have been deleted since they may not be applicable in certain cases and they might also tend to prejudice design innovations.

CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (Category B)

The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

> See comment on Criterion 6 with respect to "acceptable fuel damage limits".

CRITERION 8 - OVERALL POWER COEFFICIENT (Category B)

> We recommend deletion of this criterion since it is not applicable to certain reactor types. It is possible for the overall power coefficient resulting from a sum of components with different time constants to be positive without causing any serious safety problem. For example, in a sodium graphite reactor the coefficient has a prompt negative component together with a positive component with a long time constant. This results in an overall positive coefficient, but the negative part of the coefficient is large enough and fast enough to assure satisfactory control and safety. Safety problems relating to reactivity considerations are adequately covered in Criteria 6 and 7.

CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

> It is important to characterize the leakage as "uncontrolled". Our only other suggested change is insertion of the word, "fabricated".

CRITERION 10 - REACTOR CONTAINMENT (Category A)

Reactor containment shall be provided. The containment structure shall be designed (a) to sustain without undue risk to the health and safety of the public the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public.

> To avoid any ambiguity, "containment" should be characterized as "reactor containment". The statutory requirement of the licensee and the AEC is "to avoid undue risk to the health and safety of the public", not "to protect the public". It would

be helpful to cross reference this criterion to Criterion 37 to indicate what the AEC means by "engineered safety features". Consistent with our comments on Criterion 37, we have substituted "pipe" for "boundary" on the premise that an applicant should not be required to consider a design basis accident more conservative than the instantaneous double-ended, circumferential rupture of a large coolant pipe.

CRITERION 11 - CONTROL ROOM (Category B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposures of personnel.

As originally drafted, this criterion could be interpreted as requiring a second control room. Not only would such a requirement be inconsistent with current practice, we believe that the complexities introduced could adversely affect overall plant safety. We believe it possible to design and equip a control room to assure continuous occupancy under all circumstances, including fire. We have deleted reference to 10 CFR 20 since the radiation exposure limits set forth therein apply to normal operating conditions, not accident conditions. Compliance with the radiation exposure limits of 10 CFR 20 under accident or post-accident circumstances is neither necessary nor reasonable. We have deleted the last sentence of the original draft since it is unnecessary and contradictory with the requirement of continuous occupancy of the control room.

CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (Category B)

Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables.

We have modified this criterion to more accurately and precisely reflect its intent.

CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (Category B)

Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core.

> We have dropped the two examples since they are measures of reactivity rather than the fission process.

CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

> We have deleted the phrase "act automatically" since manual action will prove adequate, indeed desirable, in some instances.

CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (Category B)

> No change suggested.

CRITERION 16 - MONITORING REACTOR COOLANT LEAKAGE (Category B)

Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary.

> We have assumed the intent of this criterion is to assure that leakage from the primary system will be detected, not that the entire reactor coolant pressure boundary will be monitored. The latter requirement would be inconsistent with current practice and unnecessary. Also, consistent with Criterion 9, we believe that the leakage should be characterized as significant and uncontrolled.

CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (Category B)

Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive.

We believe that the modified language as indicated above more accurately and precisely reflects the intent of the criterion.

CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (Category B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels.

We believe that the modified language as indicated above more accurately and precisely reflects the intent of the criterion.

CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category B)

Protection systems shall be designed for high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public.

The suggested change is in line with our comment on Criterion 1.

CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (Category B)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served.

The significant change we have made here is to delete the last sentence of the original draft. It would appear preferable to provide duplicates of the best system or component rather than going to an inferior system or component based on a different principle.

CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

We recommend deletion of this criterion since it is more of a definition than a criterion and since the implied requirement is adequately covered by Criterion 23.

CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (Category B)

This criterion should be deleted inasmuch as its requirements, to the extent they should be included in general criteria, appear to be adequately covered by Criterion 20.

CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (Category B)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function or shall be tolerable on some other basis.

> The suggested change here includes adding to the criterion the phrase, "or shall be tolerable on some other basis".

CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (Category B)

> We recommend deletion of this criterion since it would appear preferable to focus all requirements for emergency power in Criterion 39. Note that "protection systems" has been incorporated in Criterion 39 to accommodate this deletion.

CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (Category B)

Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred.

> The reason for the changes here is that the licensee should be given some latitude in determining when and how such tests should be carried out. Further, he should be required only to test the active components of a protection system in contrast, for example, to a rupture diaphragm which could only be tested at the expense of destroying it. Also, certain tests might permit the licensee to determine if failure or loss of redundancy has occurred, but they might not permit him to demonstrate it.

CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (Category B)

> No change suggested.

CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

Two independent reactivity control systems, preferably of different principles, shall be provided.

> The phrase, "At least" which prefaced the original criterion suggests a possible escalation of requirements which we do not believe was intended.

CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition.

> Deletion of the preface phrase, "At least two of" is based on the comment made on Criterion 27. We have deleted the examples at the end of the original criterion since they could be interpreted to indicate a requirement for two fast reactivity shutdown mechanisms. This requirement is unnecessary when there is sufficient redundancy in one of the reactivity control systems to assure shutdown.

CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (Category A)

One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn.

> Deletion of the preface phrase, "At least", is consistent with the comments on Criteria 27 & 28. The other editorial changes are for purposes of clarification.

CRITERION 30 - REACTIVITY HOLDOWN CAPABILITY (Category B)

The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

> Deletion of the preface phrase, "At least one of", is consistent with the comments on Criteria 27, 28 & 29. Further, the public health and safety will not be compromised by a return to low power.

CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (Category B)

The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

> We believe the criterion should preserve its original objective and at the same time acknowledge that one of the functions of the reactor protection system is to protect against certain control system malfunctions.

CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (Category A)

Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core.

> We believe substitution of "reasonable" for "considerable" and the substitution of "lose capability of cooling the core" for "impair the effectiveness of emergency core cooling" more precisely reflects the intent of the criterion. The re-wording also correctly implies that emergency core cooling will generally be required only if the reactor coolant pressure boundary is breached.

CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (Category A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

> We have deleted the phrase, "and with only limited allowance for energy absorption through plastic deformation", on the premise that it is not helpful.

CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (Category A)

The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failures. Consideration shall be given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.

The detailed requirements contained in the original version are not appropriate for general criteria.

CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (Category A)

With the re-writing of Criterion 34 as indicated above, this criterion can and should be deleted.

CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided.

It should not be necessary to inspect or maintain surveillance over all portions of the coolant pressure boundary; hence, we have inserted the phrase, "of critical areas". We believe that both the applicant and the AEC are in a better position to take advantage of developing technology and code refinement if these general design criteria refer to "current applicable codes" rather than to specifically designated codes.

CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

> Deletion of the phrase, "As a minimum", and substitution of "piping" for "pressure boundary" are both intended to eliminate the implication that the applicant should be required to consider a design accident basis more conservative than the instantaneous, double-ended, circumferential rupture of the largest pipe in the primary system. On this premise, retention of the original language introduces a vagueness which tends to defeat the objective of the criterion.

CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (Category A)

All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public.

> Avoiding undue risk to the health and safety of the public is the purpose of all engineered safety features and the "functional reliability and ready testability" of such features is directly related to their attainment of this objective. To tie this criterion to the problem of siting appears extraneous and not helpful; hence, we have deleted the second sentence.

CRITERION 39 - EMERGENCY POWER (Category A)

An emergency power source shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems required to avoid undue risk to the health and safety of the public. This power source shall provide this capacity assuming a failure of a single active component.

> As originally drafted, this criterion could be interpreted as requiring two off-site and two on-site power sources. Since neither the AEC nor the licensee may have any control over

the off-site power supply and since an emergency on-site power supply adequate to meet the power needs of the engineered safety features is required, any reference to off-site power is irrelevant. We have, therefore, re-written this criterion to eliminate such reference to off-site power. We have also changed the title of the criterion to accommodate the addition of "protection systems", which reference was added because of the deletion of Criterion 24.

CRITERION 40 - MISSILE PROTECTION (Category A)

Adequate protection for those engineered safety features, the failure of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures.

The suggested changes in this criterion are for purposes of clarification.

CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (Category A)

Engineered safety features such as the emergency core cooling system and the containment heat removal system shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public.

We believe the measure of "sufficient performance capability" of an engineered safety feature should be that no undue risk to the public health and safety will result from the failure of any single active component of that feature. The modified language, in our opinion, more accurately and precisely reflects the intent of the criterion.

CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (Category A)

Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public.

Although it would appear extremely difficult, if not impossible, to design engineered safety features in such a way that a loss-of-coolant accident will cause no impairment of the capability of any component or system, it is possible to design them to meet the requirements of this criterion as stated above.

CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Protection against any action of the engineered safety features which would accentuate significantly the adverse after-effects of a loss of normal cooling shall be provided.

> The intent here was simply to state the criterion in a more positive way.

CRITERION 44 - EMERGENCY CORE COOLING SYSTEM CAPABILITY (Category A)

An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty.

> In our opinion, one emergency core cooling system which incorporates a sufficient redundancy of active components and covers the full range of postulated breaks should be adequate. Our modification of this criterion reflects this consensus. For this reason, we have omitted the last sentence of the original criterion.

CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM (Category A)

Design provisions shall where practical be made to facilitate physical inspection of all critical parts of the emergency core cooling system, including reactor vessel internals and water injection nozzles.

> Since inspection of water injection nozzles is not always possible on a reasonably complete and non-destructive basis and since the failure of a safety injection nozzle is assumed in most accident analyses, we have inserted the phrase, "where practical".

CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEM COMPONENTS (Category A)

> No comment other than the criterion should be presented in the context of a single emergency core cooling system, consistent with the comments offered on Criterion 44.

CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEM (Category A)

A capability shall be provided to test periodically the operability of the emergency core cooling system up to a location as close to the core as is practical.

> Testing the "operability" in contrast to the "delivery capability" of the emergency core cooling system "up to" rather than "at" a location close to the core more accurately reflects the art of the possible and should provide for as adequate a test of reliability.

CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEM (Category A)

A capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the emergency core cooling system into action, including the transfer to alternate power sources.

> The only change here, and a significant one we believe, is insertion of the word, "initially". Although we concur that a capability to test the operational sequence of the emergency core cooling system should be provided, the test as a practical matter would not be carried out frequently and possibly not more than once - prior to startup.

CRITERION 49 - REACTOR CONTAINMENT DESIGN BASIS (Category A)

The reactor containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident, including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system, will not result in undue risk to the health and safety of the public.

> The objective of this criterion, in our opinion, should be that under the circumstances of an accident the integrity of the containment should be such as to prevent

Undue risk to the health and safety of the public. Since the maintenance of containment integrity is based on effective functioning of the emergency core cooling system, it appears unreasonable in this criterion to assume the complete failure of the emergency core cooling system; hence we have assumed a failure of a single active component. Consistent with this assumption, we believe that the pressure and temperature to be withstood should be characteristic of those anticipated from the largest credible energy release associated with a loss-of-coolant accident, including the calculated energy from metal-water and other chemical reactions. Acceptance of the "failure of a single active component" concept is consistent with Criterion 41.

CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (Category A)

The selection and use of containment materials shall be in accordance with applicable engineering codes.

It appears to us that the specific requirements of this criterion as originally drafted are not in keeping with the intent of general design criteria.

CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (Category A)

If part of the reactor coolant pressure boundary is outside the containment, features shall be provided to avoid undue risk to the health and safety of the public in case of an accidental rupture in that part.

It is our understanding that it is the responsibility of the licensee to "avoid undue risk to" rather than "to protect" the health and safety of the public. We have deleted the second sentence of the criterion as originally drafted on the premise that it is only incidental to the requirement set forth in the first sentence.

CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (Category A)

Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure this system shall perform its required function, assuming failure of any single active component.

Deletion of the phrase "at least" is consistent with our comment on Criterion 27. The other changes are consistent with our comments on Criterion 41.

CRITERION 53 - CONTAINMENT ISOLATION VALVES (Category A)

No change suggested.

CRITERION 54 - INITIAL LEAKAGE RATE TESTING OF CONTAINMENT (Category A)

Containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result from the design basis accident after completion and installation of all penetrations and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance.

> We have inserted "initial" in the title to differentiate Criterion 54 from Criterion 55. Further, we believe it more realistic to leak test at peak pressures associated with postulated accidents than at design pressure. Correlation of leakage rate tests at postulated accident pressures with those conducted at design pressure prior to installation of containment penetrations will permit extrapolation of observed leakage rates to design pressure conditions.

CRITERION 55 - PERIODIC CONTAINMENT LEAKAGE RATE TESTING (Category A)

The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime.

> Our suggested changes here are consistent with our comments on Criterion 54. Further, a requirement calling for periodic leak testing at design pressure would impose an unnecessary and impractical design requirement on the plant.

CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (Category A)

Provisions shall be made to the extent practical for periodically testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at the peak pressure calculated to result from occurrence of the design basis accident.

> We have inserted the word, "periodically" to avoid an interpretation that we do not believe was intended, namely a requirement for "continuous" testing. The other suggested change is consistent with our comments on Criteria 54 & 55.

CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES (Category A)

Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

> Our only suggested change here is insertion of "to the extent practical". We believe this is consistent with the intent of the criterion as originally drafted, but we also believe that the qualification should be explicit rather than implicit. This comment also applies to Criteria 58, 59, 60, 62, 63, 64 and 65.

CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

> See comment on Criterion 57.

CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS (Category A)

> See comment on Criterion 57.

CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (Category A)

A capability shall be provided to the extent practical to test periodically the operability of the containment spray system at a position as close to the spray nozzles as is practical.

> Insertion of the phrase, "to the extent practical" is consistent with our comment on Criterion 57. The basis for substitution of "operability" for "delivery capability" is the same as that used in our comments on Criterion 47.

CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

A capability shall be provided to test initially under conditions as close as practical to the design and the full operational sequence that would bring the containment pressure-reducing systems into action, including

CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS (Category A)

See comment on Criterion 57.

CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS (Category A)

See comment on Criterion 57.

CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS (Category A)

See comment on Criterion 57.

CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (Category A)

See comment on Criterion 61.

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

No change suggested.

CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release which would result in undue risk to the health and safety of the public.

We have substituted "which would result in undue risk to the health and safety of the public" for "to plant operating areas or the public environs" since we believe the first phrase more accurately describes the responsibility of the licensee.

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category B)

Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities.

The suggested change permits the criterion to accommodate radiation limits as may be specified which may differ from those set forth in 10 CFR 20.

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity.

We have avoided the use of the word, "containment" because of its possible ambiguous connotation. The licensee may rely on some means other than containment to meet the requirements of the criterion. The other suggested changes are consistent with our comments on Criterion 67.

CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence.

We have deleted the qualification on condition (b) namely, "except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents". This qualification is not helpful and could be subject to misinterpretation by the uninformed public.

ATOMIC ENERGY COMMISSION

[10 CFR Parts 50, 115]

CODES AND STANDARDS FOR NUCLEAR POWER UNITS

Notice of Proposed Rule Making

The Atomic Energy Commission has under consideration amendments of its regulations in 10 CFR Part 50, "Licensing of Production and Utilization Facilities," and 10 CFR Part 115, "Procedures for Review of Certain Nuclear Reactors Exempted From Licensing Requirements," which would establish minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain systems and components of boiling and pressurized water-cooled nuclear power reactor units by requiring conformance with appropriate editions of published industry codes and standards.

Criterion 1 of the "General Design Criteria for Nuclear Power Plant Construction Permits" (proposed Appendix A of Part 50) [1] states that systems and components of nuclear power plants which are essential to the prevention of accidents which could affect public health and safety or to mitigation of their consequences be designed, fabricated, and tested to quality standards that reflect the importance of the safety function to be performed. It has been generally recognized that for boiling and pressurized water-cooled reactors, pressure vessels, piping, valves and pumps which are part of the reactor coolant pressure boundary should, as a minimum, be designed, fabricated, inspected, and tested in accordance with the requirements of the applicable American Society of Mechanical Engineers (ASME) codes in effect at the time the equipment is purchased, and protection systems (electrical and mechanical sensors and associated circuitry) should, as a minimum, be designed to meet the criteria developed by the Institute of Electrical and Electronics Engineers (IEEE).

The ASME codes for pressure vessels, piping, pumps, and valves and the IEEE criteria for protection systems were developed and are revised periodically by industry code committees composed of representatives of utilities, reactor designers, architect-engineers, component manufacturers, insurance companies, the Commission, and others. New industry codes and revisions to existing codes generally do not become effective for at least a year after publication for trial use and comment, and only then for contracts entered into after the effective date. Because of the time delays between the execution of the contract for and start of design or fabrication of some reactor components, 2 years may elapse between the effective dates of new or revised codes and the application of their requirements to the design and fabrication of components. Even after components complying with these code requirements are fabricated, another 2 or 3 years may elapse before the reactor is operated. The effect of this traditional pattern is that the results of currently available improved codes will not be seen in operating reactors for many years hence.

Because of the safety significance of uniform early compliance by the nuclear industry with the requirements of these ASME and IEEE codes and published code revisions, the Commission is considering the adoption of amendments to Parts 50 and 115 to require that certain components of water-cooled reactors important to safety comply with these codes and appropriate revisions to the codes at the earliest feasible time. In such reactors for which construction permits have been issued but which have not been licensed for operation, such components would be required to comply with the codes in effect at the time the equipment was ordered. In reactors for which construction permits are issued on or after April 1, 1970, such components, regardless of order date, would be required to comply with the more recent revisions of the codes (excluding Code Cases) specified in the proposed amendments.

The various dates given in the proposed amendments for compliance with the new industry codes and standards have been selected to give approximately 3 months notice of the Commission's intent to require compliance, as a condition of licensing, with specified codes or addenda that now have been available to the industry for at least 6 months. In cases where the design or fabrication of some reactor components has proceeded to the point where compliance with the specified requirements, or portions thereof, would result in hardships or unusual difficulties without a compensating increase in the level of safety, the Commission would be authorized under § 50.55a(b)(1) to grant exceptions. It should also be noted that § 50.55a(b)(2) would permit the Commission to authorize deviations from the requirements of the specified codes and standards if it can be shown that an equivalent level of safety will be provided.

The Commission considers that a significant improvement in the level of quality in design, fabrication and testing of systems and components important to safety of each reactor will be afforded by compliance with the requirements of the more recent codes specified in the proposed amendments, or portions thereof, and encourages such compliance whenever practicable, regardless of the date of purchase of equipment or the provisions of these proposed amendments. Compliance with the provisions of the proposed amendments and the referenced codes is intended to insure a basic sound quality level. It may be that the special safety importance of a particular system or component will call for supplementary measures. If analysis of the system shows that such is the case, appropriate supplementary measures are expected to be adopted by applicants and licensees, or will be required by the Commission.

Pursuant to the Atomic Energy Act of 1954, as amended, and section 553 of title 5 of the United States Code, notice is hereby given that adoption of the following amendments to 10 CFR Parts 50 and 115 is contemplated. All interested persons who desire to submit written comments or suggestions for consideration in connection with the proposed amendments should send them to the Secretary, U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Chief, Public Proceedings Branch, within 60 days after publication of the notice in the FEDERAL REGISTER. Comments received after that period will be considered if it is practicable to do so, but assurance of consideration cannot be given except as to comments filed within the period specified. Copies of comments received may be examined at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

1. Paragraph (c) of § 50.55 is amended to read as follows:

§ 50.55 Conditions of construction permits.

Each construction permit shall be subject to the following terms and conditions:

* * * * *

(c) Except as modified by this section and § 50.55a, the construction permit shall be subject to the same conditions to which a license is subject.

* * * * *

2. A new § 50.55a is added to 10 CFR Part 50 to read as follows:

§ 50.55a Codes and standards.

Each construction permit for a utilization facility shall be subject to the following conditions, in addition to those specified in § 50.55:

[1] The General Design Criteria were published for public comment in the FEDERAL REGISTER on July 11, 1967 (32 F.R. 10213).

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(a) Structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

(b) As a minimum, the systems and components of boiling and pressurized water-cooled nuclear power reactors specified in paragraphs (c), (d), (e), and (f) of this section shall meet the requirements described in those paragraphs and the protection systems of nuclear power reactors of all types shall meet the requirements described in paragraph (g) of this section, except as authorized by the Commission upon demonstration by the applicant for or holder of a construction permit that:

(1) Design, fabrication, erection, testing, or inspection of the specified system or component is, to the maximum extent practical, in accordance with generally recognized codes and standards and has proceeded to a point prior to____________ * such that compliance with the described requirements or portions thereof would result in hardships or unusual difficulties without a compensating increase in the level of safety; or

(2) Proposed deviations from the described requirements or portions thereof will be compensated for by factors or design features which provide at least an equivalent level of safety.

(c) *Pressure vessels.* For construction permits issued before April 1, 1970, for reactors not licensed for operation, pressure vessels which are part of the reactor coolant pressure boundary shall meet the requirements set forth in Section III of the American Society of Mechanical Engineers (hereinafter referred to as ASME) Boiler and Pressure Vessel Code, Applicable Code Cases, and Addenda [1] in effect at the time the vessel was ordered. For construction permits issued on or after April 1, 1970, pressure vessels which are part of the reactor coolant pressure boundary shall meet the requirements for Class A vessels set forth in the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code (excluding Code Cases), the Summer 1968 Addenda and the Winter 1968 Addenda dated June 30, and December 31, 1968, respectively, and the Summer 1969 Addenda dated June 30, 1969.[1]

(d) *Piping.* For construction permits issued before April 1, 1970, for reactors not licensed for operation, piping, and fittings which are part of the reactor coolant pressure boundary shall, if ordered before July 26, 1967, meet the requirements set forth in the American Standard Code for Pressure Piping (ASA B31.1—1955), applicable Code Cases and Addenda in effect at the time the piping or fitting was ordered, and the requirements set forth in ASA B31 Code Cases N7, N9, and N10 [1] or if ordered after July 26, 1967, meet the requirements set forth in the Power Piping Section of the USA Standard Code for Pressure Piping (USAS B31.1.0—1967) applicable Code Cases, and Addenda in effect at the time the piping or fitting was ordered, and the requirements set forth in ASA B31 Code Cases N7, N9, and N10.[1] For construction permits issued on or after April 1, 1970, piping and fittings which are part of the reactor coolant pressure boundary shall meet the requirements for Class I piping set forth in the draft Nuclear Power Piping Section of the USA Standard Code for Pressure Piping (USAS B31.7), dated February 1968 (excluding Code Cases), and Errata dated June 1968, the requirements set forth in Appendix IX—Quality Control and Nondestructive Examination Methods, of the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code, and the requirements set forth in paragraph N–153 in the Summer 1969 Addenda dated June 30, 1969, to the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code.[1]

(e) *Pumps and valves.* For construction permits issued before April 1, 1970, for reactors not licensed for operation, pumps which are part of the reactor coolant pressure boundary shall meet the nondestructive testing requirements set forth in ASA B31 Code Cases N7, N9, and N10.[1] Valves which are part of the reactor coolant pressure boundary shall if ordered before July 26, 1967, meet the requirements set forth in the American Standard Code for Pressure Piping (ASA B31.1—1955), applicable Code Cases and Addenda in effect at the time the valve was ordered, and the requirements set forth in ASA B31 Code Cases N2, N7, N9, and N10 [1] or if ordered after July 26, 1967, meet the requirements set forth in the Power Piping Section of the USA Standard Code for Pressure Piping (USAS B31.1.0—1967), applicable Code Cases, and Addenda in effect at the time the valve was ordered, and the requirements set forth in the ASA B31 Code Cases N2, N7, N9, and N10.[1] For construction permits issued on or after April 1, 1970, pumps and valves which are part of the reactor coolant pressure boundary shall meet the requirements for Class I pumps and valves set forth in the draft ASME Standard Code for Pumps and Valves for Nuclear Power, dated November 1968 (excluding Code Cases), the requirements set forth in Appendix IX—Quality Control and Nondestructive Examination Methods, of the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code, and the requirements set forth in paragraph N–153 in the Summer 1969 Addenda dated June 30, 1969 to the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code.[1]

(f) *Inservice inspection requirements.* For construction permits issued on or after April 1, 1970, pressure vessels, piping, fitting, pumps, and valves which are part of the reactor coolant pressure boundary shall meet the requirements set forth in the draft ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems, dated October 1968 (excluding Code Cases). The requirements of this paragraph need not be met by pressure-containing components whose rupture would not result in a loss of reactor coolant in excess of the replenishment capability and capacity of the normal makeup systems for the interval of time necessary to permit a reactor shutdown and orderly cooldown.

(g) *Protection systems.* For construction permits issued after April 1, 1970, protection systems shall meet the requirements set forth in the 1968 Edition of the Proposed Institute of Electrical and Electronics Engineers Criteria for Nuclear Power Plant Protection Systems (IEEE No. 279), dated August 1968.[2]

(h) *Reactor coolant pressure boundary.* As used in paragraphs (c), (d), (e), and (f) of this section, "reactor coolant pressure boundary" means all those pressure-containing components, such as pressure vessels, piping, pumps, and valves, within the following systems or portions of systems of boiling and pressurized water-cooled nuclear power reactors:

(1) The reactor coolant system. For a nuclear power reactor of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valves capable of external actuation* in the main steam and feedwater piping, and the reactor coolant system safety and relief valves.

(2) Portions of associated auxiliary systems connected to the reactor coolant system. For piping of these systems which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside the containment capable of external actuation.* For piping of these systems which contains two valves, both of which are normally closed during normal reactor operation, the boundary extends to and includes the second of these valves (the second of which must be capable of external actuation*), whether or not the system piping penetrates primary reactor containment.

(3) Portions of the emergency core cooling system connected to the reactor coolant system. For piping of this system which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside containment capable of external actuation.* For piping of this system which does not penetrate primary reactor containment, the boundary extends to and includes the second of two valves normally closed during normal reactor operation.

3. Paragraph (a) of § 115.43 is amended to read as follows:

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* Effective date of these amendments.

[1] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. Copies are available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

[2] A copy may be obtained from the Institute of Electrical and Electronics Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. A copy is available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

*Simple check valves are not acceptable for this purpose.

§ 115.43 Conditions of construction authorizations.

Each construction authorization shall be subject to the following terms and conditions.

(a) Except as modified by this section and § 115.43a, the construction authorization shall be subject to the same conditions to which an operating authorization is subject.

* * * * *

4. A new § 115.43a is added to 10 CFR Part 115 to read as follows:

§ 115.43a Codes and standards.

Each construction authorization shall be subject to the following conditions, in addition to those specified in § 115.43:

(a) Structures, systems, and components of nuclear reactors shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance to the safety function to be performed.

(b) As a minimum, the systems and components of boiling and pressurized water-cooled nuclear power reactors specified in paragraphs (c), (d), (e), and (f) of this section shall meet the requirements described in those paragraphs and the protection systems of nuclear power reactors of all types shall meet the requirements described in paragraph (g) of this section, except as authorized by the Commission upon demonstration by the applicant for or holder of a construction authorization that:

(1) Design, fabrication, erection, testing, or inspection of the specified system or component is, to the maximum extent practical, in accordance with generally recognized codes and standards and has proceeded to a point prior to ----------* such that compliance with the described requirements or portions thereof would result in hardships or unusual difficulties without a compensating increase in the level of safety; or

(2) Proposed deviations from the described requirements or portions thereof will be compensated for by factors or design features which provide at least an equivalent level of safety.

(c) *Pressure vessels.* For construction authorizations issued before April 1, 1970, for reactors not authorized for operation, pressure vessels which are part of the reactor coolant pressure boundary shall meet the requirements set forth in Section III of the American Society of Mechanical Engineers (hereinafter referred to as ASME) Boiler and Pressure Vessel Code, applicable Code Cases, and Addenda[1] in effect at the time the vessel was ordered. For construction authorizations issued on or after April 1, 1970, pressure vessels which are part of the reactor coolant pressure boundary shall meet the requirements for Class A vessels set forth in the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code (excluding Code Cases), the Summer 1968 Addenda and the Winter 1968 Addenda dated June 30, 1968, and December 31, 1968, respectively, and the Summer 1969 Addenda dated June 30, 1969.[1]

(d) *Piping.* For construction authorizations issued before April 1, 1970, piping and fittings which are part of the reactor coolant pressure boundary shall if ordered before July 26, 1967, meet the requirements set forth in the American Standard Code for Pressure Piping (ASA B31.1—1955), applicable Code Cases and Addenda in effect at the time the piping or fitting was ordered, and the requirements set forth in ASA B31 Code Cases N7, N9, and N10[1] or if ordered after July 26, 1967, meet the requirements set forth in the Power Piping Section of the USA Standard Code for Pressure Piping (USAS B31.1.0—1967), applicable Code Cases and Addenda in effect at the time the piping or fitting was ordered, and the requirements set forth in ASA B31 Code Cases N7, N9, and N10.[1] For construction authorizations issued on or after April 1, 1970, piping and fittings which are part of the reactor coolant pressure boundary shall meet the requirements for Class I piping set forth in the draft Nuclear Power Piping Section of the USA Standard Code for Pressure Piping (USAS B31.7), dated February 1968 (excluding Code Cases) and Errata dated June 1968, the requirements set forth in Appendix IX—Quality Control and Non-destructive Examination Methods, of the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code, and the requirements set forth in paragraphs N-153 in the Summer 1969 Addenda dated June 30, 1969, to the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code.[1]

(e) *Pumps and valves.* For construction authorizations issued before April 1, 1970, for reactors not authorized for operation, pumps which are part of the reactor coolant pressure boundary shall meet the nondestructive testing requirements set forth in ASA B31 Code Cases N7, N9, and N10.[1] Valves which are part of the reactor coolant pressure boundary shall if ordered before July 26, 1967, meet the requirements set forth in the American Standard Code for Pressure Piping (ASA B31.1—1955), applicable Code Cases and Addenda in effect at the time the valve was ordered, and the requirements set forth in ASA B31 Code Cases N2, N7, N9, and N10[1] or if ordered after July 26, 1967, meet the requirements set forth in the Power Piping Section of the USAS Standard Code for Pressure Piping (USAS B31.1.0—1967, applicable Code Cases, and Addenda in effect at the time the valve was ordered, and the requirements set forth in ASA B31 Code Cases N2, N7, N9, and N10.[1] For construction authorizations issued on or after April 1, 1970, pumps and valves which are part of the reactor coolant pressure boundary shall meet the requirements for Class I pumps and valves set forth in the draft ASME Standard Code for Pumps and Valves for Nuclear Power, dated November 1968 (excluding Code Cases), the requirements set forth in Appendix IX—Quality Control and Nondestructive Examination Methods, of the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code, and the requirements set forth in paragraph N-153 in the Summer 1969 Addenda dated June 30, 1969, to the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code.[1]

(f) *Inservice inspection requirements.* For construction authorizations issued on or after April 1, 1970, pressure vessels, piping, fittings, pumps, and valves which are part of the reactor coolant pressure boundary shall meet the requirements set forth in the draft ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems, dated October 1968[1] (excluding Code Cases). The requirements of this paragraph need not be met by pressure-containing components whose rupture would not result in a loss of reactor coolant in excess of the replenishment capability and capacity of the normal makeup systems for the interval of time necessary to permit a reactor shutdown and orderly cooldown.

(g) *Protection systems.* For construction authorizations issued after April 1, 1970, protection systems shall meet the requirements set forth in the 1968 Edition of the Proposed Institute of Electrical and Electronics Engineers Criteria for Nuclear Power Plant Protection Systems (IEEE No. 279), dated August 1968.[2]

(h) *Reactor coolant pressure boundary.* As used in paragraphs (c), (d), (e), and (f) of this section "reactor coolant pressure boundary" means all those pressure-containing components, such as pressure vessels, piping, pumps, and valves, within the following systems or portions of systems of boiling and pressurized water-cooled nuclear power reactors:

(1) The reactor coolant system. For a nuclear power reactor of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valves capable of external actuation,* in the main steam and feedwater piping, and the reactor coolant system safety and relief valves.

(2) Portions of associated auxiliary systems connected to the reactor coolant system. For piping of these systems which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside the containment capable of external actuation.* For piping of these

*Effective date of these amendments.

[1] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. Copies are available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

[2] A copy may be obtained from the Institute of Electrical and Electronics Engineers, United Engineering Center, 345 47th Street, New York, N.Y. 10017. A copy is available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

*Simple check valves are not acceptable for this purpose.

systems which contains two valves, both of which are normally closed during normal reactor operation, the boundary extends to and includes the second of these valves (the second of which must be capable of external actuation*), whether or not the system piping penetrates primary reactor containment.

(3) Portions of the emergency core cooling system connected to the reactor coolant system. For piping of this system which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside containment capable of external actuation.* For piping of this system which does not penetrate primary reactor containment, the boundary extends to and includes the second of two valves normally closed during normal reactor operation.

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201)

Dated at Washington, D.C., this 17th day of November, 1969.

For the Atomic Energy Commission.

W. B. McCool,
Secretary.

[F.R. Doc. 69-14004; Filed, Nov. 21, 1969; 9:34 a.m.]

INDUSTRY CODES, CODE CASES, AND ADDENDA APPLICABLE TO PRESSURE VESSELS, PIPING, VALVES, AND PUMPS WITHIN REACTOR COOLANT PRESSURE BOUNDARY

	COMPONENT PURCHASE DATE — For Nuclear Power Units With Construction Permits Issued Prior to April 1, 1970											CONSTRUCTION PERMIT DATE For Nuclear Power Units With Construction Permit Issued On or After April 1, 1970
	1965 Jan.	July	1966 Jan.	July	1967 Jan.	July	1968 Jan.	July	1969 Jan.	July	1970 Jan.	1970 April 1
SURE LS	ASME Section III 1965 Edition Subsection A Addenda Summer 1964 Applicable Code Cases	ASME Section III 1965 Edition Subsection A Applicable Code Cases	ASME Section III 1965 Edition Subsection A Addenda Summer 1965 Applicable Code Cases	ASME Section III 1965 Edition Subsection A Addenda Summer 1965 Winter 1965 Applicable Code Cases	ASME Section III 1965 Edition Subsection A Addenda Summer 1965 Winter 1965 Summer 1966 Applicable Code Cases	ASME Section III 1965 Edition Subsection A Addenda Summer 1965 Winter 1965 Summer 1966 Winter 1966 Applicable Code Cases	ASME Section III 1965 Edition Subsection A Addenda Summer 1965 Winter 1965 Summer 1966 Winter 1966 Summer 1967 Applicable Code Cases	ASME Section III 1968 Edition Subsection A Applicable Code Cases	ASME Section III 1968 Edition Subsection A Addenda Summer 1968 Applicable Code Cases	ASME Section III 1968 Edition Subsection A Addenda Summer 1968 Winter 1968 Applicable Code Cases	ASME Section III 1968 Edition Subsection A Addenda Summer 1968 Winter 1968 Summer 1969 Applicable Code Cases	ASME Section III 1968 Edition Subsection A Addenda Summer 1968 Winter 1968 Summer 1969 ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems - Oct. 1968 Draft
PIPING AND FITTINGS	ASA B31.1 1955 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N7, N9, N10	ASA B31.1 1955 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N7, N9, N10	ASA B31.1 1955 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N7, N9, N10	ASA B31.1 1955 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N7, N9, N10	ASA B31.1 1955 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N7, N9, N10	USAS B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N7, N9, N10	USAS B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N7, N9, N10	USAS B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N7, N9, N10	USAS B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N7, N9, N10	USAS B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N7, N9, N10	USAS B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N7, N9, N10	USAS B31.7 - Feb. 1968 Draft - Subsection 1, and Errata June 1968 ASME Boiler and Pressure Vessel Code - Section III - Paragraph N153 in Summer 1969 Addenda ASME Section III - Appendix IX ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems - 10/68 Draft
LVES CLUDES AFETY AND LIEF VES)	ASA B31.1 1955 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N2, N7, N9, N10	ASA B31.1 1955 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N2, N7, N9, N10	ASA B31.1 1955 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N2, N7, N9, N10	ASA B31.1 1955 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N2, N7, N9, N10	ASA B31 Code 1955 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N2, N7, N9, N10	USAS B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N2, N7, N9, N10	USAS B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N2, N7, N9, N10	USAS B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N2, N7, N9, N10	USAS B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N2, N7, N9, N10	USAS B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N2, N7, N9, N10	USA B31.1.0 1967 Edition Applicable Code Cases and Addenda ASA B31 Code Cases N2, N7, N9, N10	ASME Standard Code for Pumps and Valves for Nuclear Power-Section A - Nov. 1968 Draft ASME Boiler and Pressure Vessel Code Section III - Paragraph N153 in Summer 1969 Addenda ASME Section III - Appendix IX ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems - 10/68 Draft
PUMPS	ASA B31 Code Cases N7, N9, N10	ASA B31 Code Cases N7, N9, N10	ASA B31 Code Cases N7, N9, N10	ASA B31 Code Cases N7, N9, N10	ASA B31 Code Cases N7, N9, N10	ASA B31 Code Cases N7, N9, N10	ASA B31 Code Cases N7, N9, N10	ASA B31 Code Cases N7, N9, N10	ASA B31 Code Cases N7, N9, N10	ASA B31 Code Cases N7, N9, N10	ASA B31 Code Cases N7, N9, N10	ASME Standard Code for Pumps and Valves for Nuclear Power-Section A - Nov. 1968 Draft ASME Boiler and Pressure Vessel Code Section III - Paragraph N153 in Summer 1969 Addenda ASME Section III - Appendix IX ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems - 10/68 Draft

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