July 23, 1997 SECY-97-159

FOR: The Commissioners

FROM: L. Joseph Callan /s/
Executive Director for Operations

SUBJECT: STAFF REQUIREMENTS MEMORANDUM (SRM) DATED FEBRUARY 21, 1997, RE: BRIEFING FOR COMMISSION ON CODES AND STANDARDS, JANUARY 22, 1997

PURPOSE:

The purpose of this SECY paper is to provide the staff's response to the referenced memorandum from the Commission.

BACKGROUND:

On January 22, 1997, the NRC staff and representatives from the American Society of Mechanical Engineers (ASME) and the Institute of Electrical and Electronics Engineers (IEEE) briefed the Commission on the use of consensus codes and standards. Various aspects of design and inservice inspection requirements for mechanical systems and components in various Sections of the ASME Code were discussed. Design requirements are contained in Sections B31.1 and B31.7 of the ASME Power Piping Code and Sections I, III, and VIII of the ASME Boiler and Pressure Vessel Code (B&PVC), while Section XI of the B&PVC contains inservice inspection requirements.

The staff reviewed the differences between new and old ASME Code Editions, as well as the potential significance of these differences. As a result of that review, the staff concluded that the newer design and construction codes do not necessarily provide a reactor design that has a greater level of safety than older codes. When used correctly, both would provide generally equivalent designs in terms of level of safety.

DISCUSSION:

In the referenced SRM, the Commission requested "that the staff identify those plants which have reactor vessels or safety-related piping systems that do not fall under the design rules of ASME Code Section III and identify the design rules or Codes that do apply, characterize the significant differences between ASME Code Section III and ANSI/ASME Codes B.31.1 and B.31.7, and describe any inspection inconsistencies that could arise from the use of these differing standards."

From Final Safety Analysis Reports, the staff has compiled plant-specific data on codes or standards that were used to design reactor vessels, reactor coolant system (RCS) piping, and remaining safety-related piping (see Attachment 1). The ASME B&PVC Section III was initially issued in the early 1960s and later revised. As a result, most (i.e., 106 of 110) reactor vessels at plants licensed to operate were designed to Section III. The remaining were designed to Sections I or VIII. Safety-related piping systems of 67 plants were designed in accordance with either USA Standard (USAS) B31.1, "Power Piping," or USAS B31.7, "Nuclear Power Piping," or a combination of them with portions of piping designed to Section III. The safety-related piping systems of the remaining 43 plants were all designed to Section III.

Attachment 2 characterizes the differences in design requirements between Sections I and VIII of the Code and Section III of the Code. In summary, Sections I and VIII did not require detailed stress analysis, fatigue evaluation, thermal stress calculations, or the quality assurance measures required by Section III. However, when Sections I and VIII were used for construction of reactor vessels (and other Class 1 vessels such as pressurizers and steam generators), they were supplemented by nuclear code cases that upgraded requirements so that their initial integrity was approximately equal to that of the vessels designed to Section III.
Both the pre-Section III vessels and the early vessels designed to Section III were designed before the ASME Inservice Inspection Code, Section XI, was issued. Therefore, these vessels may not have been designed to permit access for conducting all tests and inspections required by Section XI. For piping and systems, some pre-Section III Codes did not require a detailed evaluation of thermal stresses. Pre-Section III Codes did not include specific provisions for the increase in allowable stresses for safe shutdown earthquake or postulated pipe break loads; however, additional criteria for these loadings were a part of the FSAR.

The inspection inconsistencies that arise from the differing design standards are mainly related to accessibility issues. As noted above, vessels may not have accessibility to conduct tests and inspections in accordance with Section XI, if they were designed before Section III was revised to specifically include accessibility for inspections. Consistent with the requirements of the 10 CFR 50.55a(f)(3) and 50.55a(g)(3), plants with construction permit (CP) dates later than July 1, 1974, were specifically required to provide configuration or access for Inservice Testing (IST) and Inservice Inspection (ISI), as part of their design, as described in Section XI. Plants with earlier CP dates were not specifically required to provide such configuration or access, and it is impractical to meet all of the IST and ISI requirements of the latest edition of Section XI for some components in these plants. In these cases, in accordance with 10 CFR 50.55a(f)(5), 50.55a(f)(6), 50.55a(g)(5), and 50.55a(g)(6), licensees may submit requests for relief from the requirements considered to be impractical. The NRC staff reviews the requests and, if justified, grants approval and frequently imposes alternative tests or examinations. Proposed alternatives would be expected to provide an acceptable level of quality and safety. As required by 10 CFR 50.55a(f) and 50.55a(g), operating reactor licensees must periodically update their IST and ISI programs in accordance with the latest edition and addenda of Section XI that has been incorporated by reference into 10 CFR 50.55a. The IST and ISI requirements of the regulation are applicable to all plants, independent of the design codes or standards that may have been used, to the extent practical within the limitations of design, geometry, and materials of construction of the components.

In the referenced SRM, the Commission requested that, "The staff should discuss the applicability of the ASME Code Section III or other design/construction Codes of Record to operations, in particular, to those attributes that may not be addressed by the relevant Code requirements referenced in paragraph (f) and (g) of 10 CFR 50.55a, and address the nature of design/construction Code requirements in the context of operations and the current licensing basis of operating plants."

In general, the licensing basis Design Code of Record (DCR) (e.g., ANSI, ASME) continues to apply during plant operation. The DCR requirements applicable to operating plants fall into two categories. One category contains requirements related to repair, replacement, or modification; the other contains requirements related to tests and inspections. The DCR requirements apply during operation for design related activities when a system or component is repaired, replaced, modified or newly installed. The requirements defined in 10 CFR 50.55a(f) and (g) apply for tests and inspections.

The DCR is specified in the FSAR. The staff considers a failure to meet a DCR criterion to be a nonconforming condition that requires a corrective action. For example, if the licensee's FSAR specifies ASME Section III as the DCR, Section III design criteria would be applicable when a system or component is repaired, replaced, modified or newly installed within the scope of Section III.

Pursuant to 10 CFR 50.55a(f) and (g), the continuing performance of systems or components designed under DCR is monitored and controlled by Section XI. However, Section XI may be silent regarding monitoring certain aspects of the design. Nonconforming conditions identified as a result of a Section XI required test or inspection must be corrected as specified by Section XI in accordance with the DCR (or a later Code approved by NRC). Also if a nonconforming condition is identified by some means other than a Section XI specified test or inspection during plant operation, it must be corrected in accordance with the DCR consistent with commitments made in the plant FSAR.

In the referenced SRM the Commission stated, "Since the ASME Code is a product of a consensus process, the staff should provide its rationale for applying backfit considerations when endorsing later editions of the ASME Code. The staff should discuss its approach to performing backfit analyses for the numerous changes that are reflected in ASME Code revisions. In many cases, relaxations in one portion of the Code may be a result of consensus agreement for increased requirements in other portions of the Code. In considering the option of permitting licensees to selectively determine which requirements are applicable to their facilities, the staff should discuss how the backfit analyses consider the consensus view. Staff should also address the practicalities and implications related to the inspection and enforcement of licensee's conformance to various different Code Edition requirements."

The process of incorporating by reference the ASME Code into the NRC regulations (i.e., 10 CFR 50.55a) has been in use since the first endorsement in 1971 (36 FR 11423, published 06/12/71). This has resulted in newer editions and addenda of the Code being incorporated into the regulatory process on regular basis. Licensees are required to update their IST and ISI programs every 120 months to the version of Section XI incorporated by reference in 10 CFR 50.55a. As discussed below, the staff does not perform a backfit analysis for revisions that apply to components within the present scope of 10 CFR 50.55a.

The staff's position with regard to the backfit criteria in 10 CFR 50.109 first appeared in the regulatory analysis for a final rule published in the Federal Register on June 26, 1987 (52 FR 24015). The regulatory analysis stated "It is the opinion of the Office of the General Counsel that this amendment should not be subjected to the backfit provisions in 10 CFR 50.109. The rationale is that, (1) Section III, Division 1 [Rules For Construction of Nuclear Power Plant Components], applies only to new construction and to repair and replacements (i.e., the edition and addenda to be used in the construction of a plant are selected based upon the date of the construction permit and are not changed thereafter, except voluntarily by the licensee), (2) licensees are fully aware that 10 CFR 50.55a requires that they update their ISI program every 10 years to the latest edition and addenda of Section XI that were incorporated by reference in 10 CFR 50.55a 12 months before the start of the next inspection interval, and (3) endorsing and updating references to the ASME Code, a national consensus standard developed by the participants (including the NRC) with broad and varied interests, is consistent with both the intent and spirit of the backfit rule (i.e., NRC provides for the protection of the public health and safety, and does not unilaterally impose undue burden on applicants or licensees)." Consistent with this position, when incorporating later editions or addenda of Section XI into 10 CFR 50.55a, a backfit analysis is not performed for revisions that apply to components within the present scope of 10 CFR 50.55a; although a more general regulatory analysis is developed for every endorsement of the ASME Code.

As new subsections of the Code are issued, a backfit analysis is performed for those new Code requirements that expand the scope of the 10 CFR 50.55a (e.g., the recently published rulemaking that endorsed the new Section XI Subsection IWE, "Requirements for ISI of Metal Containments and Metallic Liners of Concrete Containments," and Subsection IWL, "Requirements for ISI of Concrete Containment Components.") A backfit analysis was performed for this rulemaking that expanded the scope of 10 CFR 50.55a to require, for the first time, inspection of metal containments and concrete containment components per Section XI. Thus, a new component inspection, such as metal and concrete containments which expand the scope of 10 CFR 50.55a, is subject to the backfit criteria, and credit is not given for the consensus process.

Because ASME Code changes sometimes consist of offsetting relaxations and increased requirements, it has been the practice of the staff to endorse complete Code editions and addenda. Two recent examples of Code requirements being relaxed in one area but increased in another are the areas of support examination and pump testing. The number of supports required to be examined was decreased; however, in a directly related change, a sampling program was instituted to concentrate the examinations on areas where known problems were being detected. For pumps, prior to 1994, tests and measurements were generally performed quarterly. In 1994, the frequency for the quarterly test was standardized but the test criteria were made less stringent. In a directly related change, more stringent pump testing and measurement were required to be performed every two years. In order to incorporate the changes in pump test requirements, revisions to numerous paragraphs and tables were required. These examples support the practice of the staff to endorse complete Code editions and addenda.

The Foreword to the ASME Code cautions that complete sections should be utilized to meet the intent of the Code. While 10 CFR 50.55a(f)(4)(iv) and (g)(4)(iv) permit the licensee to use portions of subsequent editions or addenda to the ASME Code approved for use by the NRC, such use is conditioned on meeting "all related requirements of the respective editions or addenda." This reflects the NRC's recognition that many Code provisions are interrelated and that licensees should not indiscriminately use individual provisions from later editions and addenda approved for use by the NRC.

The issue of permitting licensees to selectively determine which requirements are applicable to their facility was related to a license amendment request from Entergy Operations, Inc., which submitted a request to continue use of earlier editions and addenda of the ASME Code rather than updating to a later version as was currently required by 10 CFR 50.55a. The staff viewed this request as a potential Cost Beneficial Licensing Action (CBLA). Entergy subsequently withdrew the request.

Since the staff was in the process of revising 10 CFR 50.55a to endorse a recent Code edition, the staff began to discuss this plant-specific request as an option that could be included in the ongoing rulemaking. As the staff developed the rulemaking package, Direction Setting Issue 13 (DSI-13), "Role of Industry," and the Commission's decision on this issue as reflected in COMSECY-96-062 identified additional questions related to Codes and Standards. The questions include consideration of the consensus process and application of the current backfit rule when the staff adopts updated Codes and Standards. Issues such as how the backfit analysis considers the consensus process and inspection and enforcement practicalities will be addressed in the implementation of DSI-13, which will also involve interactions with industry groups, professional societies, technical institutes, and other stakeholders.

L. Joseph Callan
Executive Director for Operations

Attachments: As stated

Attachment

Design Codes

The design codes listed in the following table are taken from two

sources. The information for
reactor vessels is taken from NUREG-1511, "Reactor Pressure Vessel Status Report," and the
information for safety-related piping is taken from the FSARs. Code cases mentioned in the
table are defined and described in Attachment 2.

DESIGN CODES USED FOR REACTOR VESSEL AND RCS PIPING

Plant Name
Unit
CP
OL
Reactor
Vessel
RCS Piping
Remaining Safety-Related Piping

Arkansas Nuclear One
1
12/06/68
05/21/74
ASME III
USAS B31.7
USAS B31.7

Arkansas Nuclear One
2
12/06/72
09/01/78
ASME III
ASME III
ASME III

Beaver Valley
1
06/26/70
07/02/76
ASME III

ANSI B31.1
ANSI B31.1

Beaver Valley
2
05/03/74
08/14/87
ASME III
ASME III

ASME III

Big Rock Point
1
05/31/60
05/01/64
ASME I, Code
Cases 1270N,
1271N, and
1273N
ASA B31.1
ASA B31.1

Braidwood
1
12/31/75
07/02/87
ASME III
ASME III
ASME III

Braidwood
2
12/31/75
05/20/88
ASME III
ASME III
ASME III

Browns Ferry
1
05/10/67
12/20/73
ASME III

USAS B31.1
USAS B31.1

Browns Ferry
2
05/10/67
08/02/74
ASME III

USAS B31.1
USAS B31.1

Browns Ferry

3
07/31/68
08/18/76
ASME III
USAS B31.1
USAS B31.1

Brunswick
1
02/07/70
11/12/76
ASME III

USAS B31.1
USAS B31.1

Brunswick
2
02/07/70
12/27/74
ASME III

USAS B31.1
USAS B31.1

Byron
1
12/31/75
02/14/85
ASME III
ASME III
ASME III

Byron
2
12/31/75
01/30/87
ASME III
ASME III
ASME III

Callaway
1
04/16/76
10/18/84
ASME III

ASME III
ASME III

Calvert Cliffs
1
07/07/69
07/31/74
ASME III
ASME III and
USAS B31.7
USAS B31.7

Calvert Cliffs
2
07/07/69
11/30/76
ASME III
ASME III and
USAS B31.7
USAS B31.7

Catawba
1
08/07/75
01/17/85
ASME III
ASME III
ASME III

Catawba
2
08/07/75
05/15/86
ASME III
ASME III
ASME III

Clinton
1
02/24/76
04/17/87
ASME III
ASME III
ASME III

Comanche Peak
1
12/19/74
04/17/90
ASME III

ASME III
ASME III

Comanche Peak
2
12/19/74
04/06/93
ASME III

ASME III
ASME III

Cooper
1
06/04/68
01/18/74
ASME III

ASME III
USAS B31.1

Crystal River
3
09/25/68
01/28/77
ASME III
USAS B31.7
USAS B31.1

Davis-Besse
1
03/24/71
04/22/77
ASME III

ASME III and
USAS B31.7
ASME III

D. C. Cook
1
03/25/69
10/25/74
ASME III

USAS B31.1

USAS B31.1

D. C. Cook
2
03/25/69
12/23/77
ASME III

USAS B31.1

USAS B31.1

Diablo Canyon
1
04/23/68
11/02/84
ASME III

ASA B31.1
USAS B31.1

Diablo Canyon
2
12/09/70
08/26/85
ASME III

ASA B31.1
USAS B31.1

Dresden
2
01/10/66
02/20/91
ASME III
USAS B31.1
USAS B31.1

Dresden
3
10/14/66
03/02/71
ASME III
USAS B31.1
USAS B31.1

Duane Arnold
1
06/22/70
02/22/74

ASME III

USAS B31.7,
USAS B31.1 and
ASME III
ASME III and
USAS B31.7

Edwin I. Hatch
1
09/30/69
10/13/74
ASME III
ASME III
USAS B31.7

Edwin I. Hatch
2
12/27/72
06/13/78
ASME III
ASME III
ASME III

Fermi
2
09/26/72
07/15/85
ASME III
USAS B31.7
ASME III

Fort Calhoun
1
06/07/68
08/09/73
ASME III
ASA B31.1 and
USAS B31.7
USAS B31.7

Ginna
1
04/25/66
12/10/84
ASME III

ASA B31.1
ASME III and

ASA B31.1

Grand Gulf
1
09/04/74
11/01/84
ASME III

ASME III
ASME III

Haddam Neck
1
05/26/64
12/27/74
ASME VIII,
Code Cases
1270N and
1273N
ASA B31.1
ASA B31.1

H. B. Robinson
2
04/13/67
09/23/70
ASME III
ASA B31.1
AWWA Class
C.200 and ASA
B31.1

Hope Creek
1
11/04/74
07/25/86
ASME III

ASME III
ASME III

Indian Point
2
10/14/66
09/28/73
ASME III

ASA B31.1
ASA B31.1

Indian Point
3
08/13/69
04/05/76
ASME III

ASA B31.1
ASA B31.1

James A. FitzPatrick
1
05/20/70
10/17/74
ASME III

ANSI B31.1
ANSI B31.1

Joseph M. Farley
1
08/16/72
06/25/77
ASME III
ASME III
ASME III

Joseph M. Farley
2
08/16/72
03/31/81
ASME III
ASME III
ASME III

Kewaunee
1
08/06/68
12/21/73
ASME III

USAS B31.1
USAS B31.1

La Salle County
1
09/10/73
08/13/82

ASME III

ASME III
USAS B31.7 and
ASME III

La Salle County
2
09/10/73
03/23/84
ASME III

ASME III
USAS B31.7 and
ASME III

Limerick
1
06/19/74
08/08/85
ASME III
ASME III
ASME III

Limerick
2
06/19/74
08/25/89
ASME III
ASME III
ASME III

Maine Yankee
1
10/21/68
06/29/73
ASME III

ASME III
USAS B31.1

McGuire
1
02/23/73
07/08/81
ASME III

ASME III
ASME III

McGuire
2
02/23/73
05/27/83
ASME III

ASME III
ASME III

Millstone
1
05/19/66
10/31/86
ASME III

ASME I, ASA
B31.1 and
ASME III
ASA B 31.1 and
ASME III

Millstone
2
12/11/70
09/26/75
ASME III

USAS B31.7 and
ASME III
USAS B31.7,
USAS B31.1 and
ASME III

Millstone
3
08/09/74
01/31/86
ASME III

ASME III
ASME III

Monticello
1
06/19/67
01/09/81
ASME III

ASME I and
USAS B31.1
ASME III

Nine Mile Point
1
04/12/65
12/26/74
ASME I, Code
Cases 1270N
and 1273N
ASME I and ASA
B31.1
ASME I and
ASA B31.1

Nine Mile Point
2
06/24/74
07/02/87
ASME III

ASME III
ASME III

North Anna
1
02/19/71
04/01/78
ASME III
USAS B31.7
USAS B31.7

North Anna
2
02/19/71
08/21/80
ASME III

USAS B31.7
USAS B31.7

Oconee
1
11/06/67
02/06/73
ASME III
USAS B31.7

USAS B31.7

Oconee
2
11/06/67
10/06/73
ASME III
USAS B31.7
USAS B31.7

Oconee
3
11/06/67
07/19/74
ASME III
USAS B31.7
USAS B31.7

Oyster Creek
1
12/15/64
07/02/91
ASME I
ASME I and ASA
31.1
ANSI B31.1 and
ASA B31.1

Palisades
1
03/14/67
02/21/91
ASME III

ASA B31.1
ASA B31.1

Palo Verde
1
05/25/76
06/01/85
ASME III

ASME III
ASME III

Palo Verde
2
05/25/76
04/24/86
ASME III

ASME III
ASME III

Palo Verde
3
05/25/76
11/25/87
ASME III

ASME III.
ASME III

Peach Bottom
2
01/31/68
12/14/73
ASME III

USAS B31.1 and
ASME III
USAS B31.1

Peach Bottom
3
01/31/68
07/02/74
ASME III

USAS B31.1 and
ASME III
USAS B31.1

Perry
1
05/03/77
11/13/86
ASME III
ASME III
ASME III

Pilgrim
1
08/26/68

09/15/72
ASME III

ANSI B31.1 and
ASME III
ANSI B31.1, and
ASME III

Point Beach
1
07/19/67
10/05/70
ASME III
USAS B31.1
USAS B31.1

Point Beach
2
07/25/68
03/08/73
ASME III
USAS B31.1
USAS B 31.1

Prairie Island
1
06/25/68
04/05/74
ASME III
USAS B31.1
USAS B31.1

Prairie Island
2
06/25/68
10/29/74
ASME III
USAS B31.1
USAS B31.1

Quad Cities
1
02/15/67
12/14/72
ASME III

USAS B31.1
USAS B 31.1 and
ASME I

Quad Cities
2
02/15/67
12/14/72
ASME III

USAS B31.1
USAS B 31.1 and
ASME I

River Bend
1
03/25/77
11/20/85
ASME III
ASME III
ASME III

Salem
1
09/25/68
12/01/76
ASME III

USAS B31.7 and
ASME III
USAS B31.7 and
ASME III

Salem
2
09/25/68
05/20/81
ASME III

USAS B31.7 and
ASME III
USAS B31.7 and
ASME III

San Onofre
2
10/18/73
09/07/82
ASME III
ASME III
ASME III

San Onofre
3
10/18/73
09/16/83
ASME III
ASME III
ASME III

Seabrook
1
07/07/76
03/15/90
ASME III

ASME III
ASME III

Sequoyah
1
05/27/70
09/17/80
ASME III

USAS B31.1 and
ASME III
USAS B31.1 and
ASME III

Sequoyah
2
05/27/70
09/15/81
ASME III

USAS B31.1 and
ASME III
USAS B 31.1 and
ASME III

Shearon Harris
1
01/27/78
01/12/87
ASME III
ASME III
ASME III

South Texas Project

1
12/22/75
03/22/88
ASME III

ASME III
ASME III

South Texas Project
2
12/22/75
03/28/89
ASME III

ASME III
ASME III

St. Lucie
1
07/01/70
03/01/76
ASME III

USAS B31.7
USAS B31.7

St. Lucie
2
05/02/77
06/10/83
ASME III

ASME III
ASME III

Summer
1
03/21/73
11/12/82
ASME III
ASME III
ASME III and
USAS B31.1

Surry
1
06/25/68
05/25/72
ASME III

ASA B31.1
ASA B31.1

Surry
2
06/25/68
01/29/73
ASME III
ASA B31.1
ASA B31.1

Susquehanna
1
11/02/73
11/12/82
ASME III
ASME III
ASME III

Susquehanna
2
11/02/73
06/27/84
ASME III
ASME III
ASME III

Three Mile Island
1
05/18/68
04/19/74
ASME III
USAS B31.7
USAS B31.7 and
USAS B31.1

Turkey Point
3
04/27/67
07/19/72
ASME III
ASA B31.1
ASA B31.1

Turkey Point
4
04/27/67
04/10/73

ASME III
ASA B31.1
ASA B31.1

Vermont Yankee
1
12/11/67
02/28/73
ASME III
ANSI B31.1
USAS B31.1 and
ANSI B31.1

Vogtle
1
06/28/74
03/16/87
ASME III

ASME III
ASME III

Vogtle
2
06/28/74
03/31/89
ASME III

ASME III
ASME III

Washington Nuclear
2
03/19/73
04/13/84
ASME III
ASME III
ASME III

Waterford
3
11/14/74
03/16/85
ASME III
ASME III
ASME III

Watts Bar

1
01/23/73
02/7/95
ASME III
ASME III
ASME III

Wolf Creek
1
05/31/77
06/04/85
ASME III

ASME III
ASME III

Zion
1
12/26/68
10/19/73
ASME III

ASA B31.1 and
USAS B31.1
USAS B31.1

Zion
2
12/26/68
11/14/73
ASME III

ASA B31.1 and
USAS B31.1
USAS B 31.1

Attachment 2

Design Differences in Applicable Codes

Codes for Reactor Vessel Design

Early reactor vessels were designed in accordance with Section I or VIII of the Code. These codes originally were written for non-nuclear vessels used in fossil power plants. NUREG-0081, "Evaluation of the Integrity of Reactor Vessels Designed to ASME Code Sections I and/or VIII," published in June 1976 documents that only ten commercial power plant reactor vessels were designed to pre-Section III codes. Four of these ten vessels (at Big Rock Point, Haddam

Neck, Nine Mile Point 1, and Oyster Creek) are in nuclear power plants licensed to operate.
The purpose of NUREG-0081 was to establish the level of integrity of reactor vessels not designed to Section III. This report was published in response to an ACRS request that a documented review be made of the status of older vessels designed in accordance with Code Sections prior to the issuance of Section III. As documented in NUREG-0081, the results of the review were stated to be "Although these vessels were designed to ASME Code Section I and/or VIII, their design and material acceptance standards were supplemented by requirements of the Navy Code, various Nuclear Code Cases, and manufacturing specifications so that their initial integrity was approximately equal to that of the vessels designed to Section III." A Code Case is an alternative to a specific portion of the Code. The Code Cases undergo the same consensus approval process as the Code.

It should be noted that later reactor vessels were all designed in accordance with Section III of the Code. The major differences between this code and earlier ones can be summarized as follows:

Section I and VIII did not require detailed stress analysis, thermal stress calculations, or the quality assurance measures required by Section III. Section I and VIII also did not require fatigue evaluation. Both the pre-Section III vessels and the early vessels designed to Section III were designed prior to the issuance of the ASME Inservice Inspection Code, Section XI; so they are not specifically designed to permit access for conducting all the examinations that the latest editions of the Code require. Thus, there are areas in these vessels that are inaccessible to inspection. As permitted by 10 CFR 50.55a, licensees request relief from the NRC from inspections that cannot be performed.

Navy Code, "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components"

The design of pre-Section III vessels was based on a U.S. Navy Code, various early ASME Nuclear Component Code Cases, and supplementary requirements of the vessels vendor. The Navy Code was a forerunner of the first design subsection of Section III. It was primarily written for design of reactor vessels in the early Naval Reactor Program, and like Section III, the Navy Code:

1. Utilized the same stress analysis methodology that was incorporated into Section III.
2. Required calculation and classification of all stresses and applied different stress limits to different categories of stresses as Section III of the Code does.
3. Required a detailed fatigue analysis and provided rules for prevention of fatigue failure.

ASME Nuclear Code Cases

1270N - Provided "General Requirements" for Nuclear vessels:

1. Vessels would be constructed to ASME Sections I or VIII modified by the requirements of the Nuclear Code Cases.

2. The order of precedence between possible conflicting requirements of the Nuclear Code Cases and ASME Section I and VIII was established, i.e., Nuclear Code Case requirements were to have precedence.

3. Vessel purchase specifications would include additional requirements to ensure vessel integrity in the unique nuclear environment for the intended life of the vessel.

1271N - This Code Case provided guidance on selection and utilization of pressure relieving devices for use in conjunction with a radioactive working fluid. In addition, it recommended that a minimum of two such devices be utilized.

1273N - Significant requirements of this Case include:

1. Steady state thermal stresses were to be combined with primary and secondary stresses resulting from the design pressure. The resulting combination of stresses was limited to three times the allowable stress at the design temperature (similar to procedure used in ASME Section III).

2. Maximum allowable design stresses for bolting material, operating at temperatures up to 800ø F, were limited to one-third the

material yield strength at temperature (similar to ASME Section III).

3. Detailed design and inspection for both full and partial penetration pressure boundary welds were included. Similar requirements were ultimately written in ASME Section III.

Codes for Piping Design

Currently, 10 CFR 50.55a requires safety-related components to be designed and fabricated to the requirements of Section III of the ASME Boiler and Pressure Vessel Code. 10 CFR 50.55a requires reactor coolant pressure boundary components to be designed to the ASME Code requirements for Class 1 components and the remaining safety-related components to be designed to ASME Code Class 2 or 3 requirements. The primary difference between the Section III design requirements for Class 1 piping and Class 2 and 3 piping is the Section III requirement for a fatigue evaluation of the Class 1 piping which includes an evaluation of local thermal stresses. Section III of the Code generally allows higher stresses for the evaluation of Class 1 piping in conjunction with the requirement for a more detailed evaluation of the stresses and more stringent fabrication and inspection requirements. Although Section III does not require an explicit fatigue evaluation of Class 2 and 3 piping, it does provide criteria for addressing cyclic thermal expansion stresses. These stresses are caused by the restraint of free thermal expansion at rigid support locations when the piping heats up and cools down.
The Code requires the allowable stress limit for thermal expansion/contraction stresses to be reduced if the number of full-temperature cycles exceeds the specified value in the Code.

Piping at many older facilities was designed to the requirements of USA Standard Code for Pressure Piping (USAS) B31.1, "Power Piping." The first code specifically written for nuclear power plant piping, USA Standard (USAS) B31.7, "Nuclear Power Piping," was initially issued for trial use and comment in 1968 and then formally issued in 1969. The B31.7 Code required a fatigue evaluation of the Class 1 piping. It also required that the Class 2 and 3 piping meet the B31.1 Code design criteria. The design criteria for Class 1 piping in B31.7 were incorporated in Section III of the ASME Code in 1971. The design criteria for Class 2 and 3 piping were taken from the B31.1 Code and also incorporated in Section III in 1971. Therefore the design requirements for ASME Class 1 piping originate from the B31.7 Code, and the design requirements for ASME Class 2 and 3 piping originate from the B31.1 Code. Although there have been numerous changes in the details of the ASME Code piping design criteria since 1971, the basic design philosophy of higher stress limits coupled with a more detailed evaluation of the stresses and more stringent fabrication and inspection requirements for ASME Class 1 piping has not changed.

The primary difference between the current ASME Section III design criteria and the B31.1 design criteria is the ASME Section III Class 1 requirement for a fatigue evaluation, including an evaluation of local thermal stresses, in conjunction with higher allowable stress limits. As a consequence, the use of the B31.1 criteria generally resulted in piping with a greater wall thickness for a given design pressure than the use of the ASME Code Class 1 criteria would have required. However, the ASME Class 1 criteria would have required a fatigue evaluation of the piping, including an evaluation of local thermal stresses. A discussion of the differences in design criteria for piping between ASME Section III Class 1 and USAS B31.1 is contained in Attachment 2 of SECY-95-245, "Completion of the Fatigue Action Plan." In it an assessment of a sample of piping components designed to the B31.1 Code using current ASME Class 1 criteria found that the current ASME fatigue limit was met for the components designed to the B31.1 Code. The details of the sample assessment as well as a more detailed discussion of the Code criteria is contained in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." In Attachment 2 of SECY-95-245, the staff concluded that the lack of a specific fatigue analysis in the design of piping components at older plants does not constitute a significant safety concern.