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# Computer-based handbook for the licensing of light-water-reactors

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**Computer-based handbook**

**for the licensing of light-water-reactors**

by

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Signatures have been redacted for privacy

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# TABLE OF CONTENTS

## LIST OF TABLES

# LIST OF FIGURES

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# CHAPTER 1. INTRODUCTION

## Nuclear Reactor Safety Design Standard

Safety is a very important issue in our lives. It touches every part of our lives, from casual things around us like safe toys for children and household items, to quite complicated things like hospitals and power plants. Even now, new technologies have to prove their safety first before they are fully accepted.

Nuclear facilities where radioactive materials are contained strictly employ the principle "safety first"; so that an acceptable balance of risk against benefit can be obtained. Most countries have adopted nuclear safety regulations which must be complied with by designers, engineers and operators at nuclear facilities. As far as radiation protection is concerned, these regulations generally derive from the Recommendations of the International Commission on Radiological Protection (ICRP), and they rest on three basic principles [1]:

1. no practice shall be adopted unless its introduction produces a positive net benefit (justification);
2. all exposures shall be kept As Low As Reasonably Achievable (ALARA), economic and social factors being taken into account (optimization);
3. the dose equivalent to individuals shall not exceed the limits recommended for

the appropriate circumstances by the Commission.

To apply these principles in a nuclear power plant, especially the second principle, optimization, a plant designer's main consideration relates to routine operation of nuclear facilities. The designer is not specifically concerned with accidents, but, of course, s/he will include consideration of such occurrences. In designing, s/he will take all reasonably practical steps, including where necessary the provision of additional plant, to minimize the likelihood and consequences of accidents. This involves consideration of the concepts of risk and risk acceptability and a balancing of the significance of possible accident situations against the practical safety measures which could be envisaged.

The core of a nuclear power plant, where the fission of uranium or plutonium fuel takes place, contains a large amount of radioactivity which needs to be contained properly so it will not cause harm to humans and environments around the plant. Strict safety procedures are applied to ensure this containment.

Heat produced in the nuclear fuel fission is removed by the coolant. Fuel overheating can occur if the heat production rate increases, exceeding the available heat removal capacity, or if the capacity of the heat removal system decreases below that required by unforeseen circumstances. There is a clear difference between these two phenomena. The first is an unplanned excursion in the reaction rate, and the second is a loss of the coolant from the primary system which is given most attention as a "loss of coolant accident" (LOCA).

In every nuclear reactor, the fundamental design goal of reactor safety is to prevent any transient from leading to damage of the fuel, particularly breaching of the fuel cladding or melting of the fuel. To prevent such damage, two main things are required: (1) the ability to shutdown the chain reaction rapidly and dependably when required, and (2) a cooling system with enough redundancy and capacity to remove the heat generated in the reactor core.

To assure that a nuclear design is safe, the acceptance criteria are based on meeting the relevant requirements in the Code of Federal Regulations (CFR). The context in which a judgment on the "safety" of a proposed nuclear power plant is made is the review of the Safety Analysis Report which the applicant must submit before the construction of the plant and just before it is licensed [2]. The U.S. Nuclear Regulatory Commission (NRC) prescribes in detail the information and analysis that must be presented in the reports.

In general, licensing procedures will be as follows [3]: Applicants will have a Preliminary Safety Analysis Report (PSAR) prepared and will submit this report as part of the application for a construction permit. The regulatory staff will undertake a review of the specific application. This staff will send out a large number of written questions concerning the PSAR. When the review is almost completed, the staff will summarize the findings, including any reservations or unresolved issues.

The Advisory Committee on Reactor Safeguards (ACRS) will then complete its review of the application for the construction permit and submit a public letter to the NRC giving its advice and recommendations concerning the application. After the ACRS letter has been submitted, an Atomic Safety and Licensing Board (ASLB) hearing can proceed and be completed. The ACRS letter will not be a part of the ASLB record and the ACRS recommendations need not be followed by the regulatory staff or even discussed at the ASLB Hearing. An ASLB Hearing is mandatory at the construction permit stage, even if there is no intervention concerning the case.

When the construction of a facility is well along, the applicants will submit a Final Safety Analysis Report (FSAR) as part of its application for an operating license. This report will provide more detail on the actual design of the facility and will reflect changes made since the PSAR. The regulatory staff will follow a similar review procedure as it does earlier for the construction permit. When the staff believes it is ready to proceed, it will again issue a safety evaluation report. Review by the ACRS and a public ACRS report on the application for an operating license will be the next step. However, a public ASLB Hearing is required only if there is intervention into the proceedings.

The criteria and standards that are used in NRC review take a number of forms [2]. The fundamental ones, from the point of view of the NRC, are the criteria that are contained in Title 10 of the CFR, the title that specifies NRC responsibilities. Part 50 of 10 CFR specifies basic standards and requirements for licensing of production and utilization facilities; a number of general design criteria, which apply both to the routine operational characteristics of reactor systems and to their accident response; quality assurance criteria for nuclear power plants and fuel reprocessing plants; requirements for emergency plans; and many other basic safety requirements. In many cases, the NRC has formulated more detailed "Regulatory Guides," which specify methods by which the applicant can satisfy these general criteria. Although these Regulatory Guides do not carry legal force, they serve practically as regulations because they turn out to be far the most convenient way to satisfy the NRC. Finally, the Regulatory Guides, and indeed the entire safety design review, rests substantially on the voluntary standards that various professional societies have formulated for the use of their members and of industry.

NRC has published a detailed description in Standard Review Plan (SRP) of the manner in which a review will be done [20]. SRP addresses in detail what is reviewed, the basis for the review, how the review is accomplished, and the nature of the conclusions that should be reached [4]. The first major section of each Review Plan, entitled "Areas of Review", describes the scope of review. The second section, entitled "Acceptance Criteria", states the purpose and technical bases for the review. The "bases" consist of specific criteria such as Atomic Energy Commission (AEC) and Nuclear Regulatory Commission (NRC) Regulatory Guides, General Design Criteria, American Society of Mechanical Engineers (ASME) Code Requirements, Branch Technical Positions, or other criteria used in the review. The third section, the "Review Procedures", discusses how the review is accomplished. This section describes the procedures in use for reviewing and approving the systems, components, data, etc., that are described in the first section of the Review Plan, using the criteria delineated in the second section. This section is generally a step-by-step procedure that the reviewer goes through to provide reasonable verification that the applicable safety criteria have been met. The fourth section, "Evaluation Findings", presents the type of conclusions that are sought regarding the acceptability of the particular review area. The final section lists the references utilized in the review process.

To help the licensing process, which needs a lot of literature searching, a computer program is written so that important information needed can be read from this program.

## Role of Computer in Nuclear Fuel Licensing

Computers are widely used nowadays. In general, the uses of computers can be divided into two classes: for numerical computation and for information and data storage, retrieval, processing, and synthesis [5]. In this thesis, the computer is used to store and process information needed in fuel licensing using an expert system shell called KnowledgePro. KnowledgePro is a new kind of communications tool. Using hypertext and expert systems, it lets the personal computer become a medium to communicate ideas and information to others [6].

In the early seventies, while computer hardware specialists were developing microchip technology, software specialists were laying the groundwork for a breakthrough in the software area. It was not to invent smaller or faster microchips, but to develop computer programs that could in some sense think, solve problems in a way that would be considered intelligent if done by a human [7]. This concept is known as artificial intelligence (AI).

Expert systems are the fruit of AI researches. In the late seventies, AI scientists found that the problem-solving power of a program comes from the knowledge it possesses. Therefore, to make a program intelligent, it is provided with lots of high-quality, specific knowledge about some problem area, so the system will be an expert in this narrow problem area. Such a program is called an expert system.

Besides being an expert system, KnowledgePro provides a hypertext system which allows users to obtain and store more information when it is needed. For example, "Does **LOCA** occur?" is understandable for nuclear engineers. People from different fields might need more information about LOCA (loss of coolant accident). By clicking the mouse button on the hypertext, they will obtain information needed about LOCA which is given only when it is called.

The combination of hypertext and expert systems which allows the designer to store, process and supply information for users is the reason why KnowledgePro is chosen to achieve the goal of this thesis.

## Objective

To make a computer program using the KnowledgePro System which will provide information and procedures needed in the licensing of light-water-cooled power reactors (LWRs).

# CHAPTER 2. LITERATURE REVIEW

## Fuel System Design

According to Standard Review Plan for Nuclear Power Plants, NUREG-0800 [20], the fuel system consists of arrays (assemblies of bundles) of fuel rods and reactivity control rods in a nuclear reactor core. To assure that there is no release of radioactive materials from an operating nuclear reactor core, the shutdown and cooling system must be available at all time. Some failures, like limited fuel failures, such as the breach of one or a few fuel rods, are acceptable, since there is no mechanism for such failures to propagate and lead to large-scale failures. A failure will be serious when it occurs in the control or cooling systems; therefore it needs to be avoided, whether from a failure that originates in the cooling equipment or from a failure (such as an electronic control failure) that leads to a cooling failure.

When a reactor loses its coolant, large portions of the fuel can quickly lose their structural integrity. Cladding, as the first line of defense to contain fission products and actinide elements, serves to maintain fuel rod structure. Following the loss of coolant in a reactor core is the rise of cladding temperature, which might lead to the melting of fuel cladding if there is no operation of the emergency core cooling system (ECCS).

If the ECCS fails, the fuel can reach temperatures so high that it melts its way through the reactor vessel and makes its way through the concrete base of the containment building. The danger is not only from the fuel escaping from the vessel, but also because breach of the fuel cladding and rise of the fuel temperature can cause release of volatile fission products, especially iodine, cesium, and the noble gases. Because of the pressure produced in the fuel element, these radionuclides can escape to the environment.

To avoid such danger, the main consideration in licensing a reactor core, i.e., the fuel and the control and cooling systems, is to ensure that most, if not all, of the radioactive materials will be contained well in the fuel element.

Fuel elements should be designed to behave in the most favorable manner in the unlikely event of a major malfunction, such as a loss of coolant flow or an overpower type of accident [8]. Researches have been made to understand the details of fuel element changes in a reactor when normally operated or in an accident, to minimize the consequences of fuel failure, and to design fail-safe elements. A subtle effect that has safety and licensing implications is the change of $UO_2$ fuel density that can occur in early life. This densification problem initially caused the collapse of cladding and unplanned neutron streaming in LWRs. A simple expedient of prepressurizing the pins with helium stopped that effect.

Another important process in fuel elements is that when uranium or plutonium atoms fission, they produce a range of fission products, especially gases, which will influence fuel element behavior. Noble gases xenon and krypton are produced in fair abundance. These gases can behave in a variety of ways such as remaining in the fuel and joining together to form bubbles which cause the fuel to swell, or diffusing to the surface of the fuel and being released into the sealed cladding and building up pressure. In either case, the cladding experiences a steadily rising pressure, which will exceed the external coolant pressure and may cause the cladding to fail. Some of the more volatile fission products, such as iodine and cesium, migrate to the cooler regions of the fuel element, i.e., to the cladding, where they may induce stress-corrosion cracking or other degradation phenomena. The discontinuities in the fuel surfaces due to pellet-pellet interfaces and to cracks result in locally high stresses on the cladding and are favored sites for chemical reactions by fission products. The term "pellet-cladding interactions," or PCI, is used to describe these effects in LWR fuel elements that are mainly responsible for early failures.

To avoid the release of radioactive materials from fuel elements, cladding is the only barrier to fission products entering the coolant stream and it must remain intact throughout its lifetime. Cladding is restricted in thickness by neutronics and thermal stress considerations. However, it must be thick and strong enough to resist swelling and coolant and fission gas pressures. To reduce these pressures, a fuel element is designed to have an empty space above or below the fuel.

There is another consideration in fuel element design. Since fuel elements are continuously exposed to radiation, most cladding materials experience a decrease in their ductility (radiation embrittlement). In addition, voids are formed which cause differential swelling and hence bowing, and interference stresses which can damage the fuel. This encourages the development of cladding and duct materials with low swelling rates. The cladding swelling, if uniform, may be advantageous in that it reduces or totally removes the fuel- swelling stresses on the cladding because the latter is always moving away from the fuel.

Substantial efforts are expended to prevent failures of the basic reactor systems, particularly of the cooling system [2]. These efforts include conservative design of cooling systems, and continuous monitoring and inspection to ensure that the systems are operable at all times. To assure that the cooling systems are always available, the basic systems are designed with redundancy. In the case that the basic systems fail, back up systems are available. These include backup reactivity control systems and ECCS.

Reactivity in a reactor core basically can be controlled using control rods and chemical-shim control. The most common control used for quick shutdown is the control rod. In general, control rods have a strong neutron absorber like cadmium over the full length of the core, and they are used for operational control of the reactor, including load following, and for quick shutdown (to "trip" or "scram" the reactor). This capability is provided by the fact that the rods can be dropped easily into place gravitationally.

In case that control rods cannot be inserted, liquid neutron absorber (chemical-shim control) containing a boron compound may be injected into the reactor to shutdown the chain reaction. Boron does not have long-lived radioactive isotopes, so no coolant radioactivity problems arise from it. Besides that, it has been found that the dissolved boron does not increase corrosion rates and that most materials suitable in neutral or high pH water at high temperature are also suitable in a boric acid solution. In addition, it has also been shown that boron does not increase the precipitation of corrosion products which would have posed problems of heat- transfer surface fouling and radioactivity levels in deposition locations [9]. Because of these characteristics, it is safe to use boron solution in a reactor core.

ECCS is another backup system, designed to operate in the primary system in the event that the coolant inventory decreases. When a LOCA occurs due to small leaks, the ECCS should be adequate to overcome the loss. When large breaks occur, the sudden release of pressure and the high temperature of the water can result in sudden formation of steam and expulsion of the coolant from the primary system, often referred to as blowdown. Under these conditions, injection of water into the primary system can be difficult. Even more, the injected water may not reach the hot fuel, so that the temperature rise in the reactor core cannot be alleviated.

In general, a reactor will have about three ECCS, which operate in distinct and independent modes. Usually, two of the systems will operate at low pressure, the condition in which most of the coolant has been lost, and one will operate at high pressure (with lower flow rates than in at least one of the low-pressure systems). A signal indicating low coolant level or abnormal pressure is required to activate ECCS, except the "accumulator" for the pressurized water reactor (PWR), which consists of a large tank of pressurized water that is kept out of the primary system by a check valve, which opens if the primary system pressure decreases below the accumulator pressure [2]. This is called a "passive" system that does not require an emergency signal to initiate operation.

In a PWR, both the primary coolant system and the various ECCS are enclosed by the containment building. Several of its ECCS are intended to cope with a broad range of accidents, from minor leakage to a rapid loss of coolant (blowdown). The only LOCA that these systems are not designed to cope with is a catastrophic rupture of the reactor vessel, in which case there is no system to hold water [2].

When a break occurs in the primary coolant system, it will cause drops in the primary system pressure. If the pressure drops to much below 1000 psi (7 MPa), the accumulators will act with no delay to inject fluid into either the reactor vessel or the cold legs depending on break location. In case of small breaks that do not greatly reduce the pressure, high pressure injection systems will provide makeup water, which is usually injected into a hot or cold leg [2].

In a boiling water reactor (BWR), a network of systems, which depend on signals indicating low water level in the pressure vessel or high pressure in the dry well, or both, perform specific ECCS functions to cope with LOCAs. The systems include high-pressure core spray to lower the pressure within the pressure vessel, provide makeup water in the event of LOCA, and cool the fuel assemblies in the event of an uncovered core. Relief valves are used to depressurize the reactor by blowing down the vessel contents into the drywell; following this step, low-pressure core spray is used to cool the fuel assemblies. The water supply for any of these systems is the suppression pool, which is also where reactor coolant losses should flow, so that a closed loop should exist.

The acceptance criteria for emergency core cooling system for light water nuclear power reactors can be found in §50.46, 10 CFR Part 50. Also ECCS evaluation models can be found in Appendix K, 10 CFR Part 50.

## Expert System

The process of building an expert system is often called knowledge engineering [7]. This process will involve human experts, and expert-system builder, called the knowledge engineers, who will "extract" from the human experts their procedures, strategies, and rules of thumb for problem solving, and build this knowledge into the expert system, as shown in Figure 2.1 [7] below. The result is a computer program

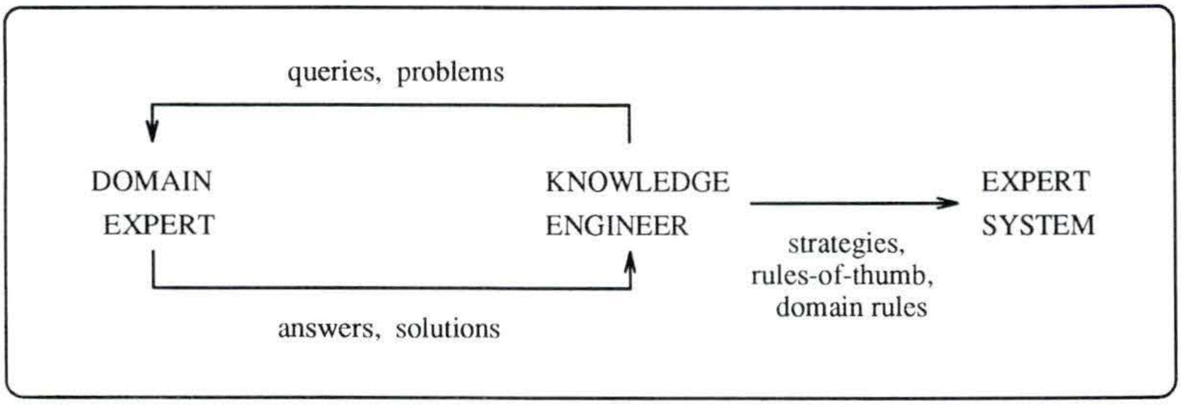


Figure 2.1: Knowledge engineering

that solves problems in much the same manner as the human experts would do.

Expert systems offer some advantages compared to human experts. They are summarized in Table 2.1 below [7]. Seeing these advantages might raise a question:

Table 2.1: The advantages of expert systems

| Human experts | Expert systems |
|---|---|
| Perishable | Permanent |
| Difficult to transfer | Easy to transfer |
| Difficult to document | Easy to document |
| Unpredictable | Consistent |
| Expensive | Affordable |

why not eliminate human experts, and replace them with expert systems? Of course there are some disadvantages of expert systems, in which human experts are clearly superior to the artificial systems. They are summarized in Table 2.2 [7].

Table 2.2: The disadvantages of expert systems

| Human experts | Expert systems |
|---|---|
| Creative | Uninspired |
| Adaptive | Needs to be told |
| Sensory experience | Symbolic input |
| Broad focus | Narrow focus |
| Common sense knowledge | Technical knowledge |

Expert systems are different from conventional programs in many ways. The basic difference is that conventional programs manipulate data, while expert systems manipulate knowledge. Thus, the conventional programs are designed to produce the correct and precise answer every time (an algorithm), and when they produce incorrect answers, the programs must be debugged. Expert systems are designed to behave like human experts, who sometimes produce correct answers and sometimes do not. Table 2.3 [21] summarizes differences between conventional programs and expert systems.

Table 2.3: The differences between conventional programs and expert systems

| Conventional programs | Expert systems |
|---|---|
| Representations and use of data | Representation and use of knowledge |
| Algorithmic | Heuristic |
| Repetitive process | Inferential process |
| Effective manipulation of large data bases | Effective manipulation of large knowledge bases |

There are some common ways to represent knowledge in expert systems: semantic networks, frames, scripts, and production systems [10].

A **semantic network** focuses on the graphical representation of relations between elements in a domain. A semantic network consists of points called nodes connected by links called arcs describing the relations between the nodes. The most common relations used in semantic networks are **isa** and **partof** relations. The 'isa' link is used to represent the fact that an element is a member of a class of elements that have a collection of distinguishing properties in common. For example, the following net represents the fact that a nuclear power plant is a type of power plant:

nuclear power plant — isa ⟶ power plants

The following net represents the fact that a control rod is a part of a nuclear power plant:

control rod — partof ⟶ nuclear power plant

These network fragments can be combined to form one net:

control rod
| partof
nuclear power plant — isa ⟶ power plants

Reasoning based on semantic nets is generally straightforward because associations can be made simply by tracing the linkages in the system [10]. For example, we can apply linkage inference to the following simple net and conclude that 63 is greater than 8:

63 — is greater ⟶ 12 — is greater ⟶ 8

A **frame** is a network of nodes and relations organized in a hierarchy, where the topmost nodes represent general concepts and the lower nodes more specific instances of those concepts [7]. It is a structure for organizing knowledge with an emphasis on default knowledge in order to imitate the important human ability to interpret new situations on the basis of knowledge gained from experience in similar situations. This ability allows our knowledge to grow with each experience rather than start from the initial conditions in every case [10]. For example, based on past experience, NucE students will expect that NucE lectures will be held in a class room in a particular building, with duration around 50 minutes, and that the class will start not more than a few minutes after the time given in the class schedule. These are students' expectations regarding NucE lectures, the thing that, unless there is evidence to the contrary, they expect to be true of all NucE lectures. We maintain large mental collections of knowledge structures that include these expectations as default values for the corresponding characteristics.

The key idea involved in frames is that our knowledge of concepts, events, and situations is organized around **expectations** of key features of those situations [11]. These expectations can be encoded in a **generic** "lecture frame" to be modified by what actually occurs during a specific lecture. This frame will include various **slots**, where specific values can be entered to describe the example above such as slots for location, equipment, etc. Some of these slots have default values at which actual values are likely to be found. Slots in one frame may contain references to other frames, thus linking them together into frame systems.

Figure 2.2 gives an illustration of how NucE Lectures and a certain lecture, i.e., NucE 551 Lecture will be presented in frames. "NucE Lecture" frame has slots for room location, start time, duration, equipment, their ranges, and their default values. This frame describes all NucE lectures in general. While "NucE 551 Lecture" frame, which describes a specific lecture, has slots with actual values only.

A **script**, which is a specialization of the general concept of a frame, is a structure that is used to store prototypes of expected sequences of events [10]. These expectations are based on our observation of recurring patterns in the events of similar situations that we have observed in the past.

## NUCE LECTURE FRAME

Specialization of: NucE Lecture

Room location:

  Range: R142, R137, R139

  Default: R142

Start time:

  Range: 0800, 0900, 1000, 1100, 1310, 1410, 1510, 1610

  Default: 0800

Duration:

  Range: 50 minutes to 90 minutes

  Default: 50 minutes

Equipment:

  Range: overhead projector, blackboard, video player, computer terminal, slide projector

  Default: blackboard, overhead projector

## NUCE 551 LECTURE

Specialization of: NucE 551 Lecture

Room location: R142

Start time: 0800

Figure 2.2: Frames for "NucE Lecture" and "NucE 551 Lecture"

Many different components can be used to construct a script. Some of the most common include:

- **Entry conditions**. The conditions that must exist for the script to be applicable.
- **Script results**. Conditions that will be true after the events in the script have occurred.
- **Props**. Slots that represent objects that are involved in the script.
- **Roles**. Slots that represent agents (e.g., people) that perform actions in the script.
- **Scenes**. Specific sequences of events that make up the script.

Figure 2.3 shows a script that represents the process of attending a class for a student.

Once we have a script that models the existing conditions, we can use the scenes to infer the existence of unobserved events. For example, considering the script of Figure 2.3, if we observe that the student left because the class was cancelled, we can infer that s/he must have been informed about that. This type of reasoning is not, however, equally reliable for predicting future events on the basis of a scene. The fact that a student is in the class does not necessarily imply that s/he will concentrate on the lecture. The student may daydream in the class.

Knowledge representation using a production system is the most popular technique. A production system is a means of codifying rule-based knowledge [11]. A rule is a piece of knowledge, one thing the program has to know [12]. It is called a rule because things to know often are of the form:

ATTENDING A CLASS SCRIPT

Props: classroom, blackboard, chalk, overhead projector, transparencies, marker pens

Roles: student, lecturer

Point of view: student

Event of sequence:

1. enter classroom
2. sit down unless the class is cancelled then exit room
3. concentrate on the lecture
4. pack up
5. exit classroom

Main concept: 3

Figure 2.3: An attending class script

if <circumstances>

then <do action, or conclude something>

Similar to human problem solving, a production system has three parts which are analogous to the knowledge states, mental operations, and decision making:

1. a working memory: a database of facts that represents what the system "knows" about the problem at any one moment;

2. a rulebase: a set of production rules that operates on those facts;

3. an interpreter that examines the current state and executes applicable production rules.

The rulebase contains the IF-THEN operations that change states of knowledge. These IF-THEN operate in a different manner from "if...then..." statements in a programming language. Each rule represents an independent "chunk" of knowledge which can be initiated, or fired, when the entire condition matches items in working memory [11]. Once the rule is fired, its actions are carried out, which usually (but not always) involves removing facts that are no longer true from working memory and adding new facts that have become true. Various actions that can be done by the rules are:

| | |
|---|---|
| make | : add a new element to global memory |
| remove | : delete an element from global memory |
| compute | : calculate a value from specified variables |
| read | : accept input from the user |
| write | : provide output to the user |

The interpreter generally operates repeatedly in the following steps:

1. Match: find the rules whose conditions are satisfied by the current contents of working memory.

2. Conflict resolution: decide which rule to use. If the condition part of none of the productions is satisfied, then halt the interpreter.

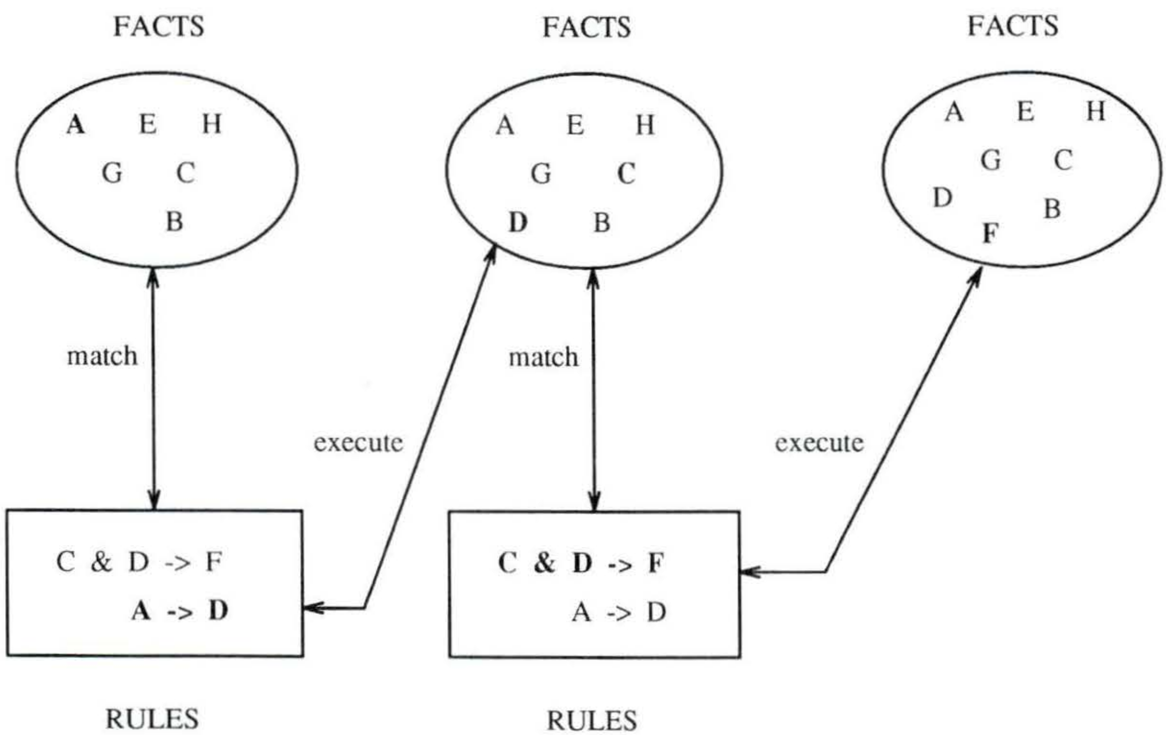


Figure 2.4: Forward chaining

3. Act: perform the actions in the rule's body.

4. Go to step 1.

There are two important ways in which rules can be used in a production system, i.e., forward chaining and backward chaining [7]. A forward chaining is an inference method where rules are matched against facts to establish new facts. Figures 2.4 and 2.5 show an example of forward chaining and its inference chain.

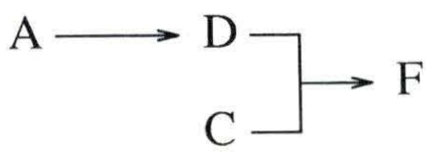

Figure 2.5: Inference chain produced by example in Figure 2.4

Figure 2.4 shows how a production system will get a final result from the system's facts and rules. The system has these rules: A $\rightarrow$ D; and C & D $\rightarrow$ F. In its working memory, the system has these facts: A, E, H, G, C, and B.

The first rule that fires is A $\rightarrow$ D because A is already in the working memory. The existence of D is inferred, and D is placed in the working memory. This causes the second rule C & D $\rightarrow$ F to fire, and now F is in the working memory. The inference chain produced is shown in Figure 2.5.

A backward chaining is an inference method where the system starts with what it wants to prove, e.g., F, and tries to establish the facts it needs to prove F. Figure 2.6 shows an example of backward chaining.

Figure 2.6 shows that in step 1 the system checks if F exists in the working memory. When it does not, then the system searches for the rule that concludes with F on the right side of the arrow, i.e., C & D $\rightarrow$ F. In step 2 till 5, the system finds C in the working memory but decides it must establish A before it can conclude D. A is found in the working memory, so D is inferred and put in the working memory.

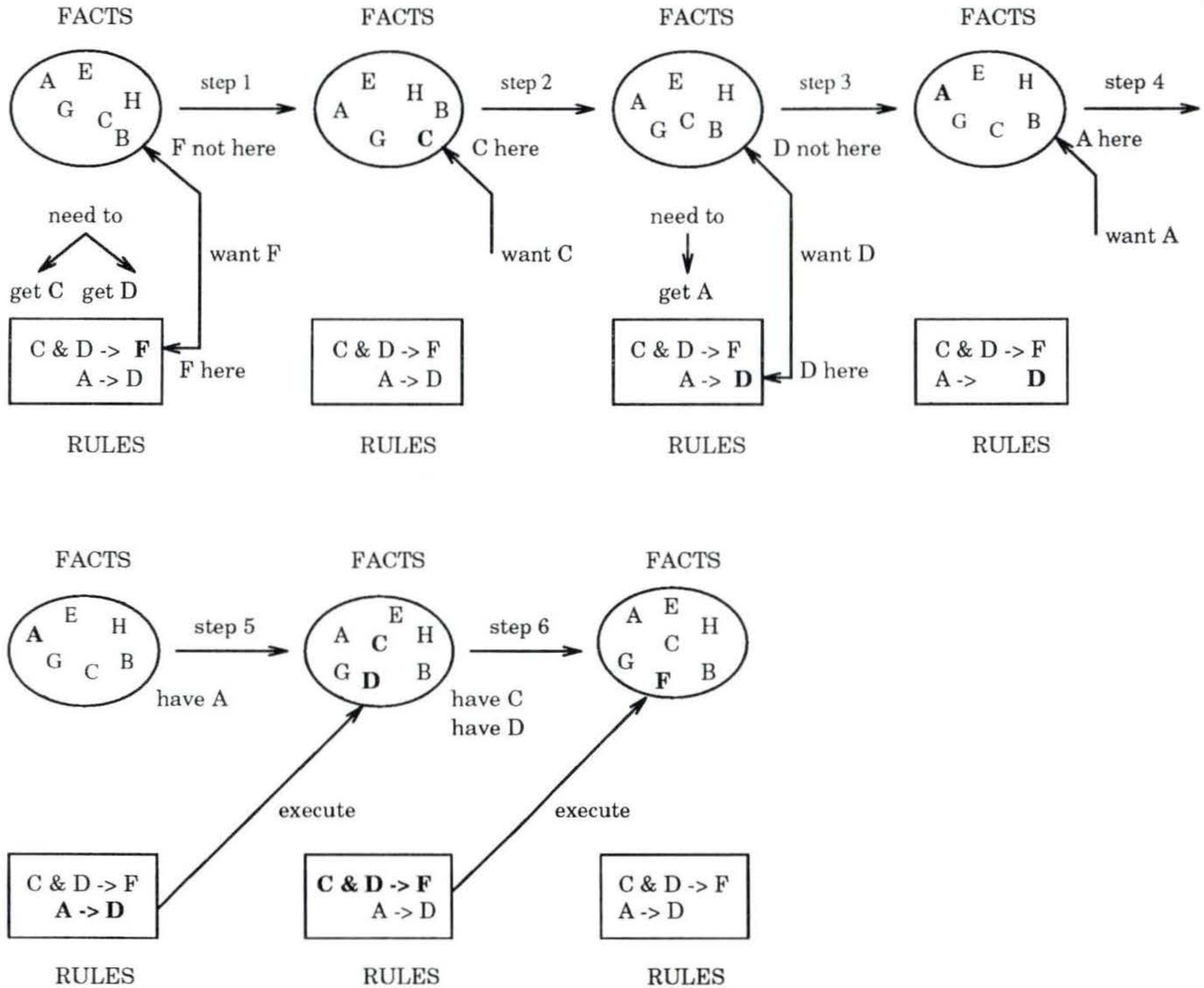


Figure 2.6: Backward chaining

Finally, since C and D are in the working memory, F is established.

KnowledgePro which is used in this thesis basically is a communication tool, or a language. It can be used in many different kinds of structures, also in a rule-based application. This is appropriate to achieve the objective of this thesis. A production system using forward chaining was used to accomplish the objective of this thesis.

# CHAPTER 3. PROGRAMMING

## Programming in KnowledgePro System

To provide information and procedures needed in the licensing of fuel for light water reactors using KnowledgePro System, variables which are called "topics" are used. A topic is a subject or a theme, a "piece of knowledge" [6].

In KnowledgePro System, topics are able to perform a variety of functions such as to:

1. contain commands like a procedure,
2. store values like a variable,
3. return values like functions,
4. be assigned properties like frames,
5. inherit values,
6. behave like system commands,
7. be threaded to hypertext,
8. be arranged hierarchically.

In this thesis, topics mostly are used to store information and contain procedure.

Topics can be defined in any order that a programmer desires. Only their level of nesting within other topics will have significance. If two topics are defined with the same name and level of nesting, only the later defined topic is used.

Since order of topics is not significant, it is important to understand how KnowledgePro searches for a topic so a program can be run effectively. A topic will be searched for:

1. in the current topic,
2. in the parent of the current topic,
3. in the siblings of the parent of the current topic,
4. in the parent of the parent,

and so on until !main is reached. (!main is a topic which is created when a KnowledgePro application is used. All other topics are descendants of !main.) Figure 3.1 will give further explanation about this search. It is important to remember that search always proceeds up and out, never down into lower hierarchical levels. This means that when siblings of parents are searched the search does not continue down inside topics belonging to the siblings. If the search is unsuccessful, the topic is either created as a child of !main or, if the search was caused by a **value_of** command, an error message is displayed saying that the topic could not be found.

From Figure 3.1, when the statement "do (ABC)", where ABC is a topic, is encountered in the topic level_3, the system will search to see if the topic already exist before it creates a new one. The topic "ABC" will be searched first in the

```
topic level_0.
        topic level_1a.
                topic level_2a.
                        topic level_3.
                        do (ABC).
                        end. (*level_3*)
                end. (*level_2a*)
                topic level_2b.
                end. (*level_2b*)
        end. (*level_1a*)
        topic level_1b.
                topic level_1ba.
                end. (*level_1ba*)
        end. (*level_1b*)
end. (*level_0*)
```

Figure 3.1: A hierarchy of topics

current topic. When it is not found, then it is searched in the parent topic, and so on until !main is reached. This order of searching is well described as follows:

level_3, level_2a, level_2b, level_1a,

level_1b, level_0, !main

Level_1ba is never examined since it is inside a sibling.

**Application**

Information provided in this program is taken from NUREG-0800 [20] Section 4.2 Fuel System Design, Section 4.3 Nuclear Design, and Section 4.4 Thermal and Hydraulic Design. Information is arranged in this program in such a way that a user can read the information through the Content or Index choice in the Opening Menu.

In general, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800 in Sections 4.2, 4.3, and 4.4 contains:

1. Areas of Review

   A. For Section 4.2 Fuel System Design, this part of NUREG- 0800 has objectives to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained.

   B. For Section 4.3 Nuclear Design, a review of the nuclear design of the fuel assemblies, control systems, and reactor core is carried out to aid in confirming that fuel design limit will not be exceeded during normal operation or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core and to assure conformance with the requirements of General Design Criteria 10-13, 20, and 25-28.

   C. For Section 4.4 Thermal and Hydraulic Design, the objectives of the review are to confirm that the thermal and hydraulic design of the core and the reactor coolant system (RCS) has been accomplished using acceptable analytical methods.

2. Acceptance Criteria

For each section, the acceptance criteria are based on meeting the relevant requirements of General Design Criteria (GDC) related to each section and also other requirements mentioned in Code of Federal Regulations (CFR) .

3. Review Procedures

The review should assure that the design bases meet the acceptance criteria.

4. Evaluation Findings

In this part, the reviewer should verify that sufficient information has been provided to satisfy the requirements of certain sections and that the evalution supports conclusions to be included in the staff's safety evaluation report.

5. Implementation

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plan for using certain sections. Except in those cases for which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

6. References

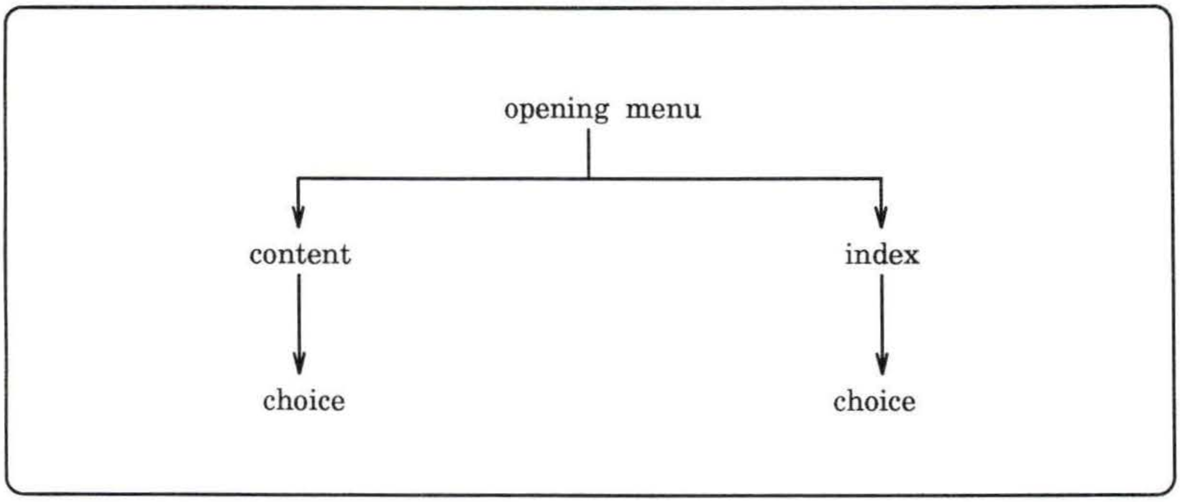


Figure 3.2: Flow-chart

To write this program, a flow-chart as shown in Figure 3.2 was used. The program will allow the user to use the procedures and information provided in NUREG-0800 step by step from the beginning following Content choice in the program, or if s/he wants to pick up a certain part or information, s/he can click on certain choices in Content or Index choice. The program is meant to be user friendly, so users without strong background in computation still can use the program easily. All information or procedures needed to use the program can be read on the screen, and users only need to click mouse's button to execute them.

As mentioned before, information and procedures needed in licensing of fuel for light water reactors will be provided in topics. To organize information so that it can be called easily, this program was designed as is shown in Figure 3.3 (this figure can

!main

Opening Menu. (which contains menu i.e., Content and Index)

Content.

Index.

Information.

Figure 3.3: Topics' organization

also be seen when the program is run using F6 key on the computer keyboard).

A topic called Opening Menu will offer a menu consisting of topic Content and Index which are two short sets of commands which function as drivers of the whole program. Information is provided in other topics which have the same level in hierarchy of topics, so when a topic is called in the children level of Content or Index, it can be found easily according to search order in KnowledgePro System.

The program was written in rule-based manner. Commands frequently used in the KnowledgePro System are:

1. ask — to ask a question
2. say — to display a message on the screen
3. if... then... — to test conditions and perform actions
4. do — to perform a topic
5. #m....#m — to begin or end marked text (hypertext).

An example of how these commands are used is given in Figure 3.4. It is from the program written for this thesis, from the "index" part to find a topic "acceptance criteria". (A complete explanation how this program works is shown in Result part of this chapter)

This part of the program is written to accommodate a user who wants to read information connected with **acceptance criteria** written in **index** part of the program. The flow is such that when the user clicks on the desired choice, in this case **acceptance criteria**, the program will respond by offering a menu containing choices of information where s/he can find this topic. Here the user can get the information s/he wants by clicking his mouse button on the choice s/he wants. After that, if s/he wants to see some more information under the same topic, there will be a short section of the program called "follow" which meets this need. This topic "follow" also contains choices to go back to Content or Index.

In NUREG-0800, the topic "acceptance criteria" can be found in topics: "4.2.II Acceptance Criteria in Fuel System Design", "4.3.II Acceptance Criteria in Nuclear Design", and "4.4.II Acceptance Criteria in Thermal and Hydraulic Design". These three topics connected with "acceptance criteria", in a production system, are located in working memory. The topic "acceptance criteria", which has some rules in it, is called the rule base. The interpreter-like component is under the topic "follow" to offer the user the follow-up steps s/he can do.

## Result

A result of this programming is a package of procedures and information based on NUREG-0800, Sections 4.2 - 4.4. To use the program, a user needs only to click

```
topic 'acceptance criteria'.
ask ('
You can find topic ''Acceptance Criteria'' in:
(please click on your choice)', click,
[
'Acceptance Criteria in Fuel System Design',
'Acceptance Criteria in Nuclear Design',
'Acceptance Criteria in Thermal and Hydraulic Design']).

if ?click is 'Acceptance Criteria in Fuel System Design'
  then do ('4.2.II Acceptance Criteria') and do (follow)
else if ?click is 'Acceptance Criteria in Nuclear Design'
  then do ('4.3.II Acceptance Criteria') and do (follow)
else if ?click is 'Acceptance Criteria in Thermal and
                  Hydraulic Design'
  then do ('4.4.II Acceptance Criteria') and do (follow).
end.


topic 'follow'.
ask ('
What would you like to see next?', next
[
'Other choices under the same topic',
'Content',
'Index']).

If ?next is 'Other choices under the same topic'
  then do (?choice)
else do (?next).
end.
```

Figure 3.4: Example of programming in KnowledgePro System

on one of the choices available on the screen.

In the beginning, a user will be offered two choices: to see the **Content** or the **Index**. This menu permits a user to get the information s/he wants. Figure 3.5 shows how this part is shown on the computer screen. In this figure, the Content

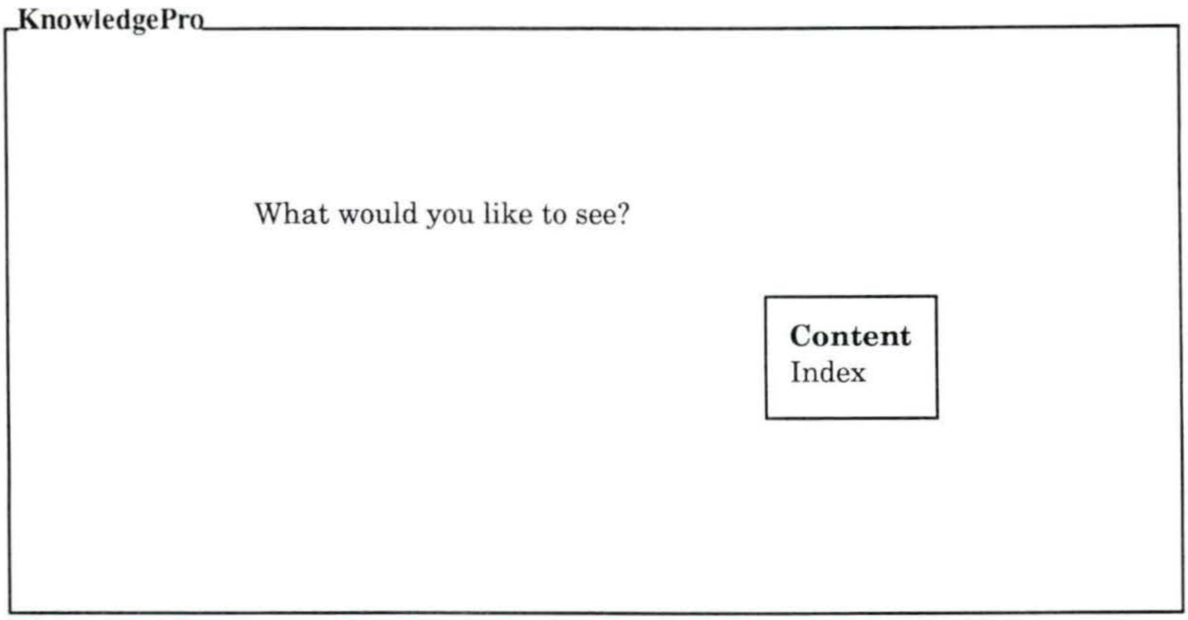


Figure 3.5: Opening menu

choice is highlighted, which means that the cursor is in this choice. To get to the choice that a user wants, s/he can use the mouse to position the cursor, or F3 key. The **F3** key lets users move the cursor forward, while **shift F3** key will bring the cursor backward. When the cursor is already on the choice that a user wants, s/he can click on the mouse button or press the **F4** key to execute the choice.

To continue the presentation of this program, it is taken that **Content** choice

**KnowledgePro**

CONTENT
(please click on your choice)

4.2 Fuel System Design
4.2.I Areas of Review
4.2.I.A Design Bases
4.2.I.B Description and Design Drawings
4.2.I.C Design Evaluation
4.2.I.D Testing, Inspection, and Surveillance Plans
4.2.II Acceptance Criteria
**4.2.II.A Design Bases**
4.2.II.A.1 Fuel System Damage
4.2.II.A.2 Fuel Rod Failure
4.2.II.A.3 Fuel Coolability

**<page 1 of 3>**

Figure 3.6: Content menu

is chosen. This step activates the topic associated with Content in the program, and presents a menu as shown in Figure 3.6.

Again here, a user can use the mouse or F3 and F4 keys to get the choice s/he wants. Since Content has a long list of choices, a user needs to use **Pg Up** and **Pg Dn** keys to see other pages of choices.

Besides the choices offered on the screen, KnowledgePro System has built-in choices which are shown in the bottom of the screen. Some of these choices are to edit, to display the hierarchical structure of the topics, to exit to DOS, to get help about how to use the system, and to quit. These choices are activated using function keys which are available on the keyboard.

A user can pick on any section s/he wants to read in the Content menu. If the user wants to read all of the material in order from Section 4.2 - 4.4 of NUREG-0800, there will be a menu at the end of any choice shown in Content menu, so that s/he does not need to go back to Content menu every time. In case the user wants to read a certain topic, and then jump to another topic, there will be a menu offering **Content** choice at the end of every part, so s/he can go back to Content to get the topic s/he wants. An example of this menu is shown in Figure 3.7, after a user clicks on the **4.2.II.A Design Bases choice**.

In this example, if a user wants to continue reading the continuation of part 4.2.II.A Design Bases, s/he can click on the choice of **4.2.II.A.1 Fuel System Damage**. Or s/he can skip that part, and go to another choice; for example, s/he can go to **Index** choice.

In **Index** choice, a user will be offered choices that will permit him to get information from selected keywords found in Sections 4.2 - 4.4 of NUREG-0800. In this example, as shown in Figure 3.8, a user was interested in getting information about **acceptance criteria**, so s/he moved the cursor to this choice in **Index** menu, and clicked on it.

After clicking on **acceptance criteria** choice, the user is offered some choices where s/he can find topic "acceptance criteria" in this program. If there is only one choice, it will be shown right away, and after that the user will be offered a menu of follow-up steps. For example, this time the user wanted to read all the information, starting from the first choice, so s/he clicked on **Acceptance Criteria in Fuel**

KnowledgePro

4.2.II.A Design Bases

The fuel system design bases must reflect the four objectives described in subsection I, Areas of Review. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. These criteria are discussed in the following:

(please click on your choice)

4.2.II.A.1 Fuel System Damage
4.2.II.A.2 Fuel Rod Failure
4.2.II.A.3 Fuel Coolability
Content
**Index**

Figure 3.7: An example of menu offered to continue reading the information

**System Design** on menu shown in Figure 3.9. Figure 3.10 shows the information shown on the screen following user's clicking on the mouse button.

If, in the example above, the user finally decided to see topic "acceptance criteria" from the menu, s/he would get into Section 4.2.II Acceptance Criteria of NUREG-0800, as it is shown in Figure 3.10. When the user is done with this reading, s/he can continue the reading by pressing Space Bar to activate a topic called "Follow" which will offer a menu as is shown in Figure 3.11.

In Figure 3.10, it also can be seen that some part of the text is highlighted. This

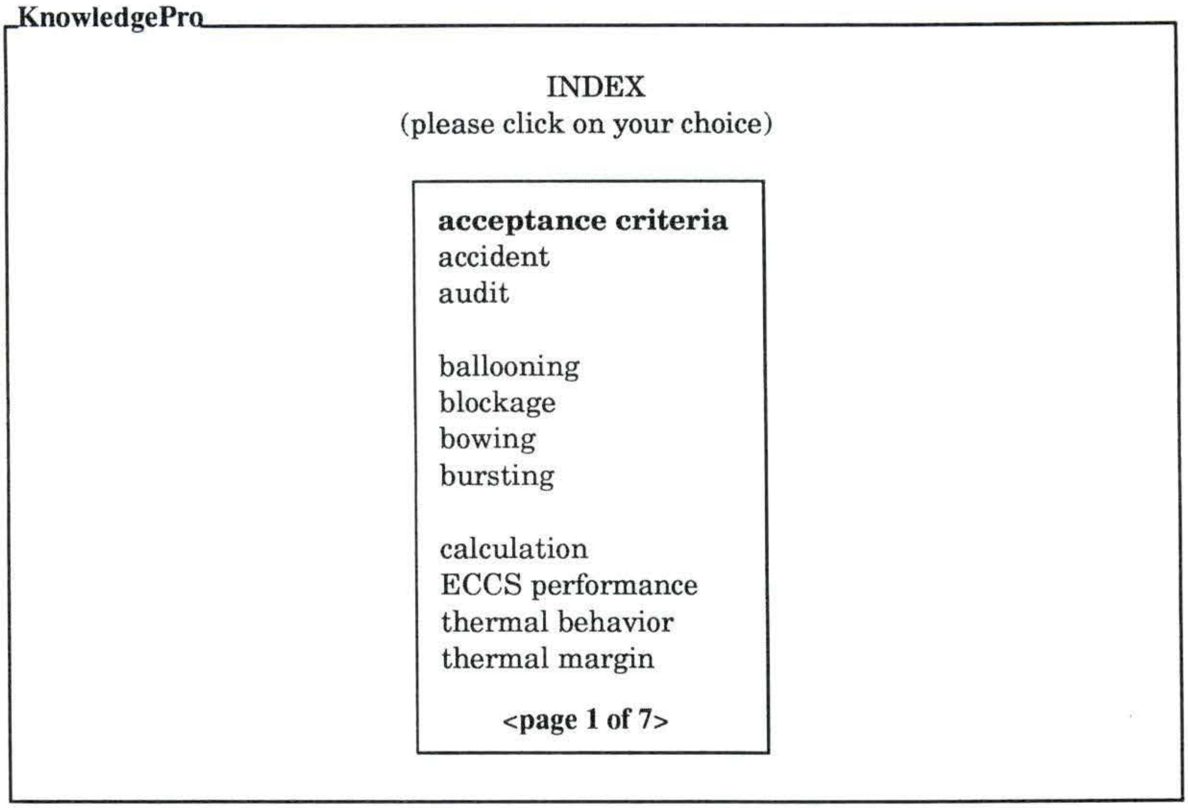


Figure 3.8: Index menu

is the one which is called "hypertext" in previous part of this thesis. The information contained in this text can be read by clicking the mouse button after positioning the cursor on the desired text. If a mouse is not available, users can use **F3** key to move the cursor forward, **Shift F3** key to move the cursor backward, and then use **F4** key to activate the hypertext.

The explanation contained in a hypertext will be written in a different window. Figure 3.12 shows what is presented on the computer screen when the user clicked

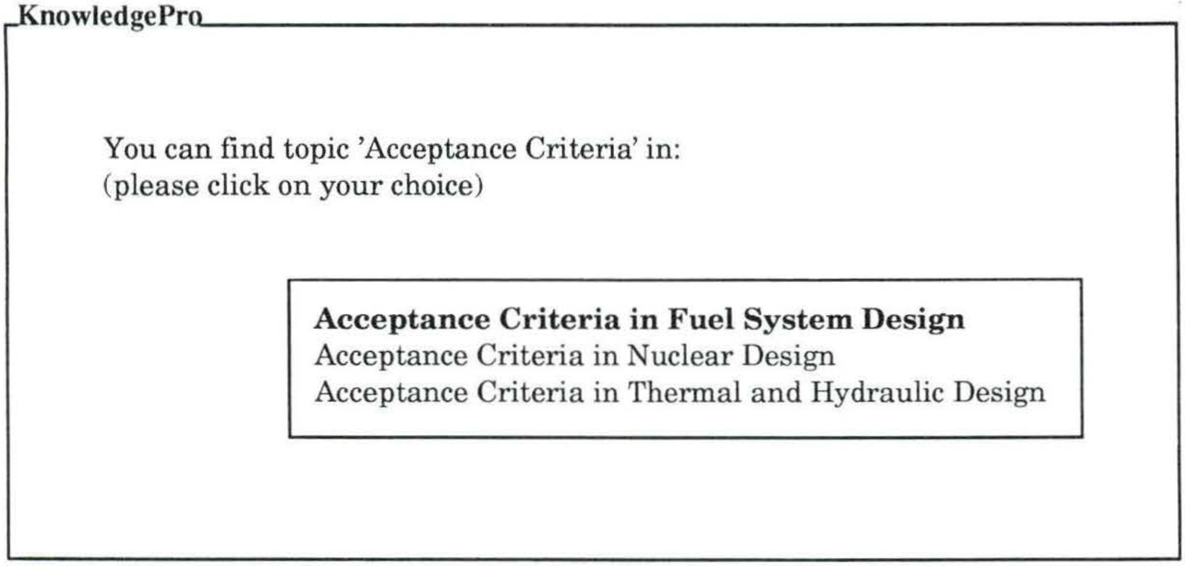


Figure 3.9: An example of finding a topic in Index menu

on **General Design Criteria 10** as it is shown in Figure 3.10.

After reading information in hypertext, when a user wants to go back to the previous window, in this case to Figure 3.10, he needs only to press "Space Bar", and the window which has information for the hypertext will disappear. When the window is not big enough to write all information it has, in the bottom of the window a note will be shown which will remind the user to use "Pg Up" and "Pg Dn" keys to get all hidden information in hypertext.

From Figure 3.11, when the **Other choice under the same topic** choice is clicked, the user will be given the menu shown in Figure 3.9, so s/he can continue reading information under the same topic, i.e., "Acceptance Criteria in Nuclear Design" and "Acceptance Criteria in Thermal and Hydraulic Design".

KnowledgePro

4.2.II Acceptance Criteria

Specific criteria necessary to meet the requirement of 10 CFR 50, **General Design Criteria 10, 17, and 35, Appendix K** to CFR Part 50; . . .

(please press Space to continue)

Figure 3.10: An example of hypertext

KnowledgePro System also permits the use of hypertext in the windows. This gives more chance for programmers to "hide" information which might interest only certain readers. When a hypertext written in a window is activated, another window will be shown on the screen to write the information contained in this hypertext. These hypertext windows will disappear one by one from the latest one when the user presses the space bar.

KnowledgePro

What would you like to see next?
(please click on your choice)

**Other choices under the same topic**
Content
Index

Figure 3.11: Menu in topic "Follow"

KnowledgePro

Specific criteria necessary to meet the requirements of 10 CFR Part 50, **General Criteria 10, 27, and 35; Appendix K** to 10 CFR Part 50;

Criterion 10 - Reactor Design

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Figure 3.12: An example of a window for hypertext

# CHAPTER 4. CONCLUSION

As mentioned in the Introduction, the objective of this thesis is to write a program using KnowledgePro System to facilitate licensing of light water reactors. This objective was accomplished in a program as provided in the Appendix B. The program was written using the KnowledgePro System shell in a forward chaining production system.

The program will benefit plant engineers and others who are interested in light water reactor licensing by providing relevant information and procedures, so they do not need to search through as many literature references as would otherwise be necessary. The program is written as a user friendly program, so even a user without any computer skill or background can use it by following the directions provided on the screen.

Though the program has provided necessary licensing information and procedures, it is not a final product that can be solely relied upon in a licensing process. A lot of information, especially technical ones, is not available in this program yet. This information can be added easily to the program, or it can be kept in a separate file and be called by the program using "load" command. Both approaches have disadvantages. If the additional information is added directly to the program, the program becomes big, and more difficult to debug. On the other hand, this approach has one distinct advantage, i.e., easy access to information so the information can be obtained right away when it is needed. If the information is kept in a separate file, it has the advantage of being easier to program and to debug. But it will take time to access the information and present it to the user.

If this program is to be developed to be a complete guide for licensing, it is necessary to communicate with the NRC so the information and procedures in the program will be current. Besides that, the program needs a safety feature so that its contents cannot be altered by an unauthorized person.

It is beneficial to write a program in KnowledgePro System, because the commands and the logic are simple. The only disadvantage found while programming in KnowledgePro System was its inability to edit programs larger than 150 kbytes. Many times the system failed to save such a file and the final result of editing could not be saved. To solve this problem the program was written in Wordstar as a non-document file, then loaded to KnowledgePro System. Another solution would be to break the program into some smaller programs. This method has the disadvantage of longer execution time, and so it was not chosen for this project.

Although programming in KnowledgePro System is easy because the commands are used in daily language (ask, say, do, etc.), it has to be done carefully. When a notation like ",", ".", or "/" is missed, it will adversely affect the whole program. These mistakes sometimes cannot be found easily because the error message given during compilation of the program does not show the exact location of the mistake. Moreover, if the programmer ignores the error message, the compilation will continue, and the program might be able to execute, but it will not give the correct result.

One important thing to remember in KnowledgePro System is the hierarchy of the topics. This is because of the way KnowledgePro System searches a topic. It is important to make sure that a desired topic is not hidden in an unaccessible location. Another important thing is not to give the same name to two or more topics in the same level of nesting, because only the latter defined topic is used.

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# APPENDIX A. PROGRAM MANUAL

To start using The Licensing of Fuel for Light-water-cooled Power Reactors Program, a user should follow this procedure:

1. change the directory into KnowledgePro directory (in Computer Laboratory of Nuclear Engineering Laboratory it is done by choosing KnowledgePro in network).

2. type KP to start KnowledgePro System.

3. choose topic 'License' in the KnowledgePro System choices by moving cursor to this choice, and then click your mouse button on your choice.

4. follow the instruction in the program to get the wanted information.

# APPENDIX B. COMPUTER PROGRAM

```
do ('Opening Menu').

topic 'Opening Menu'.
ask ('
                         Computer-based Handbook
                for the Licensing of Light Water Reactors

                                   by

                      Maria Adiartsi Hendrawarsita


                       under the supervision of

                        Dr. Monroe S. Wechsler


                        Iowa State University
                             Ames, Iowa

This program has been developed to provide information and procedures
needed in the licensing of light-water-cooled power reactors (LWRs).

Sources of Information:

   1. Standard Review Plan for the Review of Safety Analysis Reports
      for Nuclear Power Plants: LWR Edition. USNRC Report NUREG-0800.
      1981.
   2. 10 CFR Part 50, Appendix A. General Design Criteria for Nuclear
      Power Plants.
   3. 10 CFR Part 50, Appendix K. ECCS Evaluation Models.
   4. 10 CFR Part 50, & 50.46. Acceptance Criteria for Emergency Core
      Cooling Systems for Light Water Nuclear Power Reactors.
   5. 10 CFR Part 100. Reactor Site Criteria.
```

```
This program is menu based and to search through this you can use
either a mouse or keys on your keyboard.

If you are using a mouse, position the cursor on the page count at the
bottom of the screen and use the left button to move up a page and the
right button to move down a page. To select a choice from the menu, you
can just move the cursor to the choice and click on the left button. You
can also do this with the Hypertext and the system menu at the bottom of
the screen. Hypertext contains more information on a highlighted topic
and this is displayed when the topic is selected.

When you do not have a mouse, use the arrow keys to highlight the choice
in the menu and hit the Enter/Return key to execute. If you are viewing
information extending over a screenful, use the PgUp/PgDn keys to read the
entire section. To select any of the system menus, hit the appropriate
function key as shown at the bottom of the screen. To select Hypertext,
use F3 key to move from one highlighted text to the next. Shift-F3 will
take you back to the previous highlighted text. To execute a particular
Hypertext selection use the F4 key.


What would you like to see?', want,
     [
Content,
Index,
'Opening Menu']).
do (?want).

topic 'Content'.
ask ('

                                CONTENT
                      (please click on your choice)',choice,
[
'4.2 Fuel System Design',
'4.2.I Areas of Review',
'4.2.I.A Design Bases',
'4.2.I.B Description and Design Drawings',
'4.2.I.C Design Evaluation',
'4.2.I.D Testing, Inspection, and Surveillance Plans',
'4.2.II Acceptance Criteria',
'4.2.II.A Design Bases',
'4.2.II.A.1 Fuel System Damage',
'4.2.II.A.2 Fuel Rod Failure',
'4.2.II.A.3 Fuel Coolability',
'4.2.II.B Description and Design Drawings',
'4.2.II.C Design Evaluation',
'4.2.II.C.1 Operating Experience',
```

```
'4.2.II.C.2 Prototype Testing',
'4.2.II.C.3 Analytical Predictions',
'4.2.II.D Testing, Inspection, and Surveillance Plans',
'4.2.II.D.1 Testing and Inspection of New Fuel',
'4.2.II.D.2 On-line Fuel System Monitoring',
'4.2.II.D.3 Post-irradiation Surveillance',
'4.2.III Review Procedures',
'4.2.IV Evaluation Findings',
'4.2.V Implementation',
'4.2.VI References',
'4.2.VII Appendix',
'

4.3 Nuclear Design',
'4.3.I Areas of Review',
'4.3.II Acceptance Criteria',
'4.3.III Review Procedures',
'4.3.IV Evaluation Finding',
'4.3.V Implementation',
'4.3.VI References',
'4.3.VII Appendix',
'

4.4 Thermal and Hydraulic Design',
'4.4.I Areas of Review',
'4.4.II Acceptance Criteria',
'4.4.III Review Procedures',
'4.4.IV Evaluation Findings',
'4.4.V Implementation',
'4.4.VI References']).
do (?choice).
end.

topic 'index'.
ask ('
                                                INDEX
                         (please click on your choice)', choice,
[
'acceptance criteria',
'analysis',
'analytical predictions',
'
ballooning',
'blockage',
'bowing',
'bursting',
'
cladding',
```

```
'   embrittlement',
'   melting',
'   moisture level',
'   overheating',
'   pellet/cladding interaction',
'coolability',
'
damage',
'deformation',
'densification',
'design basis',
'
earthquake',
'
fracturing',
'fretting',
'fuel',
'   pellet overheating',
'   rod ballooning',
'   rod bowing',
'   rod failure',
'   system damage',
'   system design',
'   temperature',
'   violence expulsion',
'
grid',
'
hydraulic design',
'
loss-of-coolant-accident',
'
nuclear design',
'
overheating of',
'
stored energy']).

do (?choice).
end.

topic '4.2 Fuel System Design'.
do ('I Fuel System Design').
end.

topic '4.2.I Areas of Review'.
do ('I.A Areas of Review').
ask ('Please click on your choice', area,
```

```
[
'4.2.I.A Design Bases',
'4.2.I.B Description and Design Drawings',
'4.2.I.C Design Evaluation',
'4.2.I.D Testing, Inspection, and Surveillance Plans',
'Content',
'Index',
'Opening Menu']).

do (?area).
end.

topic '4.2.I.A Design Bases'.
do ('I.A.1 Design Bases').
ask ('Please click on your choice', next,
[
'4.2.I.B Description and Design Drawings',
'4.2.I.C Design Evaluation',
'4.2.I.D Testing, Inspection, and Surveillance Plans',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.I.B Description and Design Drawings'.
do ('I.A.2 Description and Design Drawings').
ask ('Please click on your choice', next,
[
'4.2.I.A Design Bases',
'4.2.I.C Design Evaluation',
'4.2.I.D Testing, Inspection, and Surveillance Plans',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.I.C Design Evaluation'.
do ('I.A.3 Design Evaluation').
ask ('Please click on your choice', next,
[
'4.2.I.A Design Bases',
'4.2.I.B Description and Design Drawings',
'4.2.I.D Testing, Inspection, and Surveillance Plans',
'Content',
```

```
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.I.D Testing, Inspection, and Surveillance Plans'.
do ('I.A.4 Testing, Inspection, and Surveillance Plans').
ask ('Please click on your choice', next,
[
'4.2.I.A Design Bases',
'4.2.I.B Description and Design Drawings',
'4.2.I.C Design Evaluation',
'4.2.II Acceptance Criteria',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.II Acceptance Criteria'.
do ('I.B Acceptance Criteria').
ask ('Please click on your choice', area,
[
'4.2.II.A Design Bases',
'4.2.II.B Description and Design Drawings',
'4.2.II.C Design Evaluation',
'4.2.II.D Testing, Inspection, and Surveillance Plans',
'Content',
'Index',
'Opening Menu']).

do (?area).
end.

topic '4.2.II.A Design Bases'.
do ('I.B.1 Design Bases').
ask ('Please click on your choice', next,
[
'4.2.II.A.1 Fuel System Damage',
'4.2.II.A.2 Fuel Rod Failure',
'4.2.II.A.3 Fuel Coolability',
'Content',
'Index',
'Opening Menu']).

do (?next).
```

```
end.

topic '4.2.II.A.1 Fuel System Damage'.
do ('I.B.1.a Fuel System Damage').
ask ('Please click on your choice', next,
[
'4.2.II.A.2 Fuel Rod Failure',
'4.2.II.A.3 Fuel Coolability',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.II.A.2 Fuel Rod Failure'.
do ('I.B.1.b Fuel Rod Failure').
ask ('Please click on your choice', next,
[
'4.2.II.A.1 Fuel System Damage',
'4.2.II.A.3 Fuel Coolability',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.II.A.3 Fuel Coolability'.
do ('I.B.1.c Fuel Coolability').
ask ('Please click on your choice', next,
[
'4.2.II.A.1 Fuel System Damage',
'4.2.II.A.2 Fuel Rod Failure',
'4.2.II.B Description and Design Drawings',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.II.B Description and Design Drawings'.
do ('I.B.2 Description and Design Drawings').
ask ('Please click on your choice', next,
[
'4.2.II.A Design Bases',
'4.2.II.C Design Evaluation',
```

```
'4.2.II.D Testing, Inspection, and Surveillance Plans',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.II.C Design Evaluation'.
do ('I.B.3 Design Evaluation').
ask ('Please click on your choice', next,
[
'4.2.II.C.1 Operating Experience',
'4.2.II.C.2 Prototype Testing',
'4.2.II.C.3 Analytical Predictions',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.II.C.1 Operating Experience'.
do ('I.B.3.a Operating Experience').
ask ('Please click on your choice', next,
[
'4.2.II.C.2 Prototype Testing',
'4.2.II.C.3 Analytical Predictions',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.II.C.2 Prototype Testing'.
do ('I.B.3.b Prototype Testing').
ask ('Please click on your choice', next,
[
'4.2.II.C.1 Operating Experience',
'4.2.II.C.3 Analytical Predictions',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.
```

```
topic '4.2.II.C.3 Analytical Predictions'.
do ('I.B.3.c Analytical Predictions').
ask ('Please click on your choice', next,
[
'4.2.II.C.1 Operating Experience',
'4.2.II.C.2 Prototype Testing',
'4.2.II.D Testing, Inspection, and Surveillance Plans',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.II.D Testing, Inspection, and Surveillance Plans'.
do ('I.B.4 Testing, Inspection, and Surveillance Plans').
ask ('Please click on your choice', next,
[
'4.2.II.D.1 Testing and Inspection of New Fuel',
'4.2.II.D.2 On-line Fuel System Monitoring',
'4.2.II.D.3 Post-irradiation Surveillance',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.II.D.1 Testing and Inspection of New Fuel'.
do ('I.B.4.a Testing and Inspection of New Fuel').
ask ('Please click on your choice', next,
[
'4.2.II.D.2 On-line Fuel System Monitoring',
'4.2.II.D.3 Post-irradiation Surveillance',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.II.D.2 On-line Fuel System Monitoring'.
do ('I.B.4.b On-line Fuel System Monitoring').
ask ('Please click on your choice', next,
[
'4.2.II.D.1 Testing and Inspection of New Fuel',
'4.2.II.D.3 Post-irradiation Surveillance',
'Content',
```

```
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.II.D.3 Post-irradiation Surveillance'.
do ('I.B.4.c Post-irradiation Surveillance').
ask ('Please click on your choice', next,
[
'4.2.II.D.1 Testing and Inspection of New Fuel',
'4.2.II.D.2 On-line Fuel System Monitoring',
'4.2.III Review Procedures',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.III Review Procedures'.
do ('I.C Review Procedures').
ask ('Please click on your choice', next,
[
'4.2.IV Evaluation Findings',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.IV Evaluation Findings'.
do ('I.D Evaluation Findings').
ask ('Please click on your choice', next,
[
'4.2.V Implementation',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.V Implementation'.
do ('I.E Implementation').
ask ('Please click on your choice', next,
[
```

```
'4.2.VI References',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.VI References'.
do ('I.F References').
ask ('Please click on your choice', next,
[
'4.2.VII Appendix',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.2.VII Appendix'.
do ('I.G Appendix').
ask ('Please click on your choice', next,
[
'4.2.I Areas of Review',
'4.2.II Acceptance Criteria',
'4.2.III Review Procedures',
'4.2.IV Evaluation Findings',
'4.2.V Implementation',
'4.2.VI References',
'4.3 Nuclear Design',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.3 Nuclear Design'.
do ('II Nuclear Design').
end.

topic '4.3.I Areas of Review'.
do ('II.A Areas of Review').
ask ('Please click on your choice', next,
[
'4.3.II Acceptance Criteria',
'Content',
```

```
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.3.II Acceptance Criteria'.
do ('II.B Acceptance Criteria').
ask ('Please click on your choice', next,
[
'4.3.III Review Procedures',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.3.III Review Procedures'.
do ('II.C Review Procedures').
ask ('Please click on your choice', next,
[
'4.3.IV Evaluation Finding',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.3.IV Evaluation Finding'.
do ('II.D Evaluation Finding').
ask ('Please click on your choice', next,
[
'4.3.V Implementation',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.3.V Implementation'.
do ('II.E Implementation').
ask ('Please click on your choice', next,
[
'4.3.VI References',
'Content',
```

```
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.3.VI References'.
do ('II.F References').
ask ('Please click on your choice', next,
[
'4.3.VII Appendix',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.3.VII Appendix'.
do ('II.G Appendix').
ask ('Please click on your choice', next,
[
'4.3.I Areas of Review',
'4.3.II Acceptance Criteria',
'4.3.III Review Procedures',
'4.3.IV Evaluation Finding',
'4.3.V Implementation',
'4.3.VI References',
'4.4 Thermal and Hydraulic Design',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.4 Thermal and Hydraulic Design'.
do ('III Thermal and Hydraulic Design').
end.

topic '4.4.I Areas of Review'.
do ('III.A Areas of Review').
ask ('Please click on your choice', next,
[
'4.4.II Acceptance Criteria',
'Content',
'Index',
'Opening Menu']).
```

```
do (?next).
end.

topic '4.4.II Acceptance Criteria'.
do ('III.B Acceptance Criteria').
ask ('Please click on your choice', next,
[
'4.4.III Review Procedures',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.4.III Review Procedures'.
do ('III.C Review Procedures').
ask ('Please click on your choice', next,
[
'4.4.IV Evaluation Findings',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.4.IV Evaluation Findings'.
do ('III.D Evaluation Findings').
ask ('Please click on your choice', next,
[
'4.4.V Implementation',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic '4.4.V Implementation'.
do ('III.E Implementation').
ask ('Please click on your choice', next,
[
'4.4.VI References',
'Content',
'Index',
'Opening Menu']).
```

```
do (?next).
end.

topic '4.4.VI References'.
do ('III.F References').
ask ('Please click on your choice', next,
[
'4.4.I Areas of Review',
'4.4.II Acceptance Criteria',
'4.4.III Review Procedures',
'4.4.IV Evaluation Findings',
'4.4.V Implementation',
'Content',
'Index',
'Opening Menu']).

do (?next).
end.

topic 'acceptance criteria'.
ask ('
You can find topic ''Acceptance Criteria'' in:
(please click on your choice)', click,
[
'Acceptance Criteria in Fuel System Design',
'Acceptance Criteria in Nuclear Design',
'Acceptance Criteria in Thermal and Hydraulic Design']).

if ?click is 'Acceptance Criteria in Fuel System Design'
   then do ('I.B Acceptance Criteria') and do (follow)
else if ?click is 'Acceptance Criteria in Nuclear Design'
   then do ('II.B Acceptance Criteria') and do (follow)
else if ?click is 'Acceptance Criteria in Thermal and Hydraulic Design'
   then do ('III.B Acceptance Criteria') and do (follow).
end.

topic 'follow'.
ask ('
What would you like to do next?', see,
[
'Other choices under the same topic',
'Content',
'Index',
'Opening Menu']).

if ?see is 'Other choices under the same topic' then do (?choice)
else do (?see).
```

```
end.

topic 'analysis'.
say ('
You can find topic ''Analysis of Loads'' in:
Appendix of Fuel System Design, part A
(please press Space to continue)')
and do ('4.2.VII Appendix').
end.

topic 'analytical predictions'.
say ('
You can find topic ''Analytical Predictions'' in:
4.2.II.C.3 Analytical Predictions
(please press Space to continue)')
and do ('4.2.II.C.3 Analytical Predictions').
end.

topic 'ballooning'.
ask ('
You can find topic ''Ballooning'' in:
(please click on your choice)', click,
[
'Fuel Coolability, part d',
'Analytical Predictions, part e']).

if ?click is 'Fuel Coolability, part d'
   then do ('I.B.1.c Fuel Coolability') and do (follow)
else if ?click is 'Analytical Predictions, part e'
   then do ('I.B.3.c Analytical Predictions') and do (follow).
end.

topic 'blockage'.
say ('
See topic ''Ballooning''
(please press Space to continue)')
and do ('ballooning').
end.

topic 'bowing'.
say ('
You can find topic ''Bowing'' in:
4.2.II.C.3 Analytical Predictions, part c
(please press Space to continue)')
and do ('4.2.II.C.3 Analytical Predictions').
end.

topic 'bursting'.
```

```
ask ('
You can find topic ''Bursting'' in:
(please click on your choice)', click,
[
'Fuel Rod Failure, part h',
'Fuel Coolability, part d']).

if ?click is 'Fuel Rod Failure, part h'
   then do ('I.B.1.b Fuel Rod Failure') and do (follow)
else if ?click is 'Fuel Coolability, part d'
   then do ('I.B.1.c Fuel Coolability') and do (follow).
end.

topic 'cladding'.
ask ('
You can find topic connected to ''Cladding'' in:
(please click on your choice)', click,
[
'   embrittlement',
'   melting',
'   moisture level',
'   overheating',
'   pellet/cladding interaction']).

do (?click).
end.

topic '   embrittlement'.
say ('
You can find topic ''Cladding Embrittlement'' in:
4.2.II.A.3 Fuel Coolability, part a
(please press Space to continue)')
and do ('4.2.II.A.3 Fuel Coolability').
end.

topic '   melting'.
say ('
You can find topic ''Cladding Melting'' in:
4.2.II.A.3 Fuel Coolability, part c
(please press Space to continue)')
and do ('4.2.II.A.3 Fuel Coolability').
end.

topic '   moisture level'.
say ('
You can find topic ''Cladding Moisture Level'' in:
4.2.II.A.2 Fuel Rod Failure, part a
(please press Space to continue)')
```

```
and do ('4.2.II.A.2 Fuel Rod Failure').
end.

topic '   overheating'.
say ('
You can find topic ''Overheating of Cladding'' in:
4.2.II.A.2 Fuel Rod Failure, part d
(please press Space to continue)')
and do ('4.2.II.A.2 Fuel Rod Failure').
end.

topic '   pellet/cladding interaction'.
say ('
You can find topic ''Pellet/Cladding Interaction'' in:
4.2.II.A.2 Fuel Rod Failure, part g
(please press Space to continue)')
and do ('4.2.II.A.2 Fuel Rod Failure').
end.

topic 'coolability'.
say ('
You can find topic ''Fuel Coolability'' in:
4.2.II.A.3 Fuel Coolability
(please press Space to continue)')
and do ('4.2.II.A.3 Fuel Coolability').
end.

topic 'damage'.
say ('
You can find topic ''Fuel System Damage'' in:
4.2.II.A.1 Fuel System Damage
(please press Space to continue)')
and do ('4.2.II.A.1 Fuel System Damage').
end.

topic 'deformation'.
say ('
You can find topic ''Structural Deformation'' in:
4.2.II.A.3 Fuel Coolability, part e
(please press Space to continue)')
and do ('4.2.II.A.3 Fuel Coolability').
end.

topic 'densification'.
say ('
You can find topic ''Densification Effect'' in:
4.2.II.C.3 Analytical Predictions, part b
(please press Space to continue)')
```

```
and do ('4.2.II.C.3 Analytical Predictions').
end.

topic 'design basis'.
say ('
You can find topic ''Design Basis'' in:
4.2.II.A Design Bases
(please press Space to continue)')
and do ('4.2.II.A Design Bases').
end.

topic 'earthquake'.
say ('
You can find topic ''Safe Shutdown Earthquake'' in
4.2.VII Appendix, part D.2
(please press Space to continue)')
and do ('4.2.VII Appendix').
end.

topic 'fracturing'.
say ('
You can find topic ''Mechanical Fracturing'' in
4.2.II.A.2 Fuel Rod Failure, part i
(please press Space to continue)')
and do ('4.2.II.A.2 Fuel Rod Failure').
end.

topic 'fretting'.
say ('
You can find topic ''Fretting'' in
4.2.II.A.2 Fuel Rod Failure, part c
(please press Space to continue)')
and do ('4.2.II.A.2 Fuel Rod Failure').
end.

topic 'fuel'.
ask ('
You can find topic connected to ''Fuel'' in:
(please click on your choice)', click,
[
'   pellet overheating',
'   rod ballooning',
'   rod bowing',
'   rod failure',
'   system damage',
'   system design',
'   temperature',
'   violence expulsion']).
```

```
do (?click).
end.

topic '   pellet overheating'.
say ('
You can find topic ''Overheating of Fuel Pellets'' in:
4.2.II.A.2 Fuel Rod Failure, part e
(please press Space to continue)')
and do ('4.2.II.A.2 Fuel Rod Failure').
end.

topic '   rod ballooning'.
say ('
See topic ''Ballooning''
(please press Space to continue)')
and do ('ballooning').
end.

topic '   rod bowing'.
say ('
See topic ''Bowing''
(please press Space to continue)')
and do ('bowing').
end.

topic '   rod failure'.
say ('
You can find topic ''Fuel Failure'' in:
4.2.II.A.2 Fuel Rod Failure
(please press Space to continue)')
and do ('4.2.II.A.2 Fuel Rod Failure').
end.

topic '   system damage'.
say ('
You can find topic ''Fuel System Damage'' in:
4.2.II.A.1 Fuel System Damage
(please press Space to continue)')
and do ('4.2.II.A.1 Fuel System Damage').
end.

topic '   system design'.
say ('
You can find topic ''Fuel System Design'' in:
4.2 Fuel System Design
(please press Space to continue)')
and do ('4.2 Fuel System Design').
```

```
end.

topic '   temperature'.
say ('
You can find topic ''Fuel Temperatures'' in:
4.2.II.C.3 Analytical Predictions, part a
(please press Space to continue)')
and do ('4.2.II.C.3 Analytical Predictions').
end.

topic '   violence expulsion'.
say ('
You can find topic ''Violent Expulsion of Fuel'' in:
4.2.II.A.3 Fuel Coolability, part b
(please press Space to continue)')
and do ('4.2.II.A.3 Fuel Coolability').
end.

topic 'grid'.
say ('
You can find topic ''Grids'' in
4.2.VII Appendix A, part C
(please press Space to continue)')
and do ('4.2.VII Appendix').
end.

topic 'hydraulic design'.
say ('
You can find topic ''Hydraulic Design'' in:
4.4 Thermal and Hydraulic Design
(please press Space to continue)')
and do ('4.4 Thermal and Hydraulic Design').
end.

topic 'loss-of-coolant-accident'.
say ('
You can find topic ''Loss-of-coolant-accident'' in:
4.2.VII Appendix A, part D
(please press Space to continue)')
and do ('4.2.VII Appendix').
end.

topic 'nuclear design'.
say ('
You can find topic ''Nuclear Design'' in:
4.3 Nuclear Design
(please press Space to continue)')
and do ('4.3 Nuclear Design').
```

```
end.

topic 'overheating of'.
say ('
You can find topic ''Overheating of'' in:
1. Fuel Rod Failure, part d, i.e., Overheating of Cladding, and
2. Fuel Rod Failure, part e, i.e., Overheating of Fuel Pellets
(please press Space to continue)')
and do ('4.2.II.A.2 Fuel Rod Failure').
end.

topic 'stored energy'.
say ('
See topic ''Fuel Temperatures''
(please press Space to continue)')
and do ('temperature').
end.

topic 'I Fuel System Design'.
ask ('
4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITY

Primary - Core Performance Branch (CPB)
Secondary - None

This part contains :
(click on your choice)',part,
[
'4.2.I Areas of Review',
'4.2.II Acceptance Criteria',
'4.2.III Review Procedures',
'4.2.IV Evaluation Findings',
'4.2.V Implementation',
'4.2.VI References',
'4.2.VII Appendix',
'Content',
'Index',
'Opening Menu']).

do (?part).
end.

topic 'I.A Areas of Review'.
say ('
4.2.I Areas of Review
```

The thermal, mechanical, and materials design of the fuel system is evaluated by CPB. The fuel system consists of arrays (assemblies or bundles) of fuel rods including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; spacer grids and springs; end plates; channel boxes; and reactivity control rods. In the case of the control rods, this section covers the reactivity control elements that extend from the coupling interface of the control rod drive mechanism into the core. The Mechanical Engineering Branch reviews the design of control rod drive mechanism in Standard Review Plan (SRP) Section 3.9.4 and the design of reactor internals in SRP Section 3.9.5.

The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged," as used in the above statement, means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design #mCriterion 10#m (Ref.1), and the design limts that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. fuel rod failures must be accounted for in the dose analysis required by #m10 CFR Part 100#m (Ref. 2) for postulated accidents. "Coolability," in general, means that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the General Design Criteria (e.g., #mGDC 27 and 35#m). Specific coolability requirements for the loss-of-coolant accident are given in 10 CFR Part 50, #m&50.46#m (Ref. 3).

All fuel damage criteria are described in SRP Section 4.2. For those criteria that involve DNBR or CPR limits, specific thermal-hydraulic criteria are given in #mSRP Section 4.4#m. The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to the Accident Evaluation Branch for use in estimating the radiological consequences of plant releases.

The fuel system review covers the following specific areas.

(please press Space to continue)').

```
topic '10 CFR Part 100'.
text is read ('p100.ws').
write (con:,?text).
```

```
do ('I.A Areas of Review').
end.

topic '&50.46'.
text is read ('p50_46.ws').
write (con:,?text).
do ('I.A Areas of Review').
end.

end.

topic 'I.A.1 Design Bases'.
say ('
4.2.I.A Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and
provide limiting values for important parameters such that damage will be
limited to acceptable levels. The design bases should reflect the safety
review objectives as described above.

(please press Space to continue)').
end.

topic 'I.A.2 Description and Design Drawings'.
say ('
4.2.I.B Description and Design Drawings

The fuel system description and design drawing are reviewed. In general, the
description will emphasize product specifications rather than process speci-
fications.

(please press Space to continue)').
end.

topic 'I.A.3 Design Evaluation'.
say ('
4.2.I.C Design Evaluation

The performance of the fuel system during normal operation, anticipated opera-
tional occurrences, and postulated accidents is reviewed to determine if all
design bases are met. The fuel system components, as listed above, are revi-
ewed not only as separate components but also as integral units such as fuel
rods and fuel assemblies. The review consists of an evaluation of operating
experience, direct experimental comparisons, detailed mathematical analyses,
and other information.

(please press Space to continue)').
end.
```

```
topic 'I.A.4 Testing, Inspection, and Surveillance Plans'.
say ('
4.2.I.D Testing, Inspection, and Surveillance Plans

Testing and inspection of new fuel is performed by the licensee to ensure that
the fuel is fabricated in accordance with the design and that it reaches the
plant site and is loaded in the core without damage. On-line fuel rod failure
monitoring and postirradiation surveillance should be performed to detect ano-
malies or confirm that the fuel system is performing as expected; surveillance
of control rods containing B4C should be performed to ensure against reacti-
vity loss. The testing, inspection, and surveillance plans along with their
reporting provisions are reviewed by CPB to ensure that the important fuel
design considerations have been addressed.

(please press Space to continue)').
end.


topic 'I.B Acceptance Criteria'.
say ('
4.2.II Acceptance Criteria

Specific criteria necessary to meet the requirements of 10 CFR Part 50,
#m&50.46#m; #mGeneral Design Criteria 10, 27, and 35#m; #mAppendix K#m to 10
CFR Part 50; and #m10 CFR Part 100#m identified in subsection I of this SRP
section as follows:

(please press Space to continue)').

topic '&50.46'.
text is read ('p50_46.ws').
write (con:,?text).
do ('I.B Acceptance Criteria').
end.

topic 'Appendix K'.
text is read ('app_k.ws').
write (con:,?text).
do ('I.B Acceptance Criteria').
end.

topic '10 CFR Part 100'.
text is read ('p100.ws').
write (con:,?text).
do ('I.B Acceptance Criteria').
end.
```

```
end.

topic 'I.B.1 Design Bases'.
say ('
4.2.II.A Design Bases

The fuel system design bases must reflect the four objectives described in
subsection I, Areas of Review. To satisfy these objectives, acceptance crite-
ria are needed for fuel system damage, fuel rod failure, and fuel coolability.
These criteria are discussed in the following:

(please press Space to continue)').
end.

topic 'I.B.1.a Fuel System Damage'.
say ('
4.2.II.A.1 Fuel System Damage

This subsection applies to normal operation, and the information to be revi-
ewed should be contained in Section 4.2 of the Safety Analysis Report.

To meet the requirements of General Design #mCriterion 10#m as it relates to
Specified Acceptable Fuel Design Limits for normal operation, including anti-
cipated operational occurrences, fuel system damage criteria should be given
for all known damage mechanisms.

Fuel system damage includes fuel rod failure, which is discussed below in sub-
section II.A.2. In addition to precluding fuel rod failure, fuel damage crite-
ria should assure that fuel system dimensions remain within operational tole-
rances and that functional capabilities are not reduced below those assumed in
the safety analysis. Such damage criteria should address the following to be
complete.

(a) Stress, strain, or loading limits for spacer grids, guide tubes, thimbles,
    fuel rods, control rods, channel boxes, and other fuel system structural
    members should be provided. Stress limits that are obtained by methods
    similar to those given in Section III of the ASME Code (Ref. 4) are accep-
    table. Other proposed limits must be justified.

(b) The cumulative number of strain fatigue cycles on the structural members
    mentioned in paragraph (a) above should be significantly less than the
    design fatigue lifetime, which is based on appropriate data and includes a
    safety factor of 2 on stress amplitude or a safety factor of 20 on the
    number of cycles (Ref. 5). Other proposed limits must be justified.

(c) Fretting wear at contact points on the structural members mentioned in
    paragraph (a) above should be limited. The allowable fretting wear should
    be stated in the Safety Analysis Report and the stress and fatigue limits
```

in paragraph (a) and (b) above should presume the existence of this wear.

(d) Oxidation, hydriding, and the buildup of corrosion products (crud) should be limited. Allowable oxidation, hydriding, and crud levels should be discussed in the Safety Analysis Report and shown to be acceptable. These levels should be presumed to exist in paragraph (a) and (b) above. The effect of crud on thermal-hydraulic considerations is reviewed as described in SRP Section 4.4.

(e) Dimensional changes such as rod bowing or irradiation growth of fuel rods, control rods, and guide tubes need be limited to set values (i.e., damage limits), but they must be included in the design analysis to establish operational tolerances.

(f) Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation unless otherwise justified.

(g) Worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly (either gravity or holddown springs). Hydraulic loads for this evaluation are reviewed as described in SRP Section 4.4.

(h) Control rod reactivity must be maintained. This may require the control rods to remain watertight if water-soluble or leachable materials (e.g., B4C) are used.

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(please press Space to continue)').
end.

topic 'I.B.1.b Fuel Rod Failure'.
say ('
4.2.II.A.2 Fuel Rod Failure
```

This subsection applies to normal operation, anticipated operational occurrences, and postulated accidents. Paragraph (a) through (c) address failure mechanisms that are more limiting during normal operation, and the information to be reviewed should be contained in Section 4.2 of the Safety Analysis Report. Paragraph (d) through (h) address failure mechanisms that are more limiting during anticipated operational occurrences and postulated accidents, and the information to be reviewed will usually be contained in Chapter 15 of the Safety Analysis Report. Paragraph (i) should be addressed in Section 4.2 of the Safety Analysis Report because it is not addressed elsewhere.

To meet the requirements of (a) General Design #mCriterion 10#m as it relates to Specified Acceptable Fuel Design Limits for normal Operation, including anticipated operational occurrences, and (b) #m10 CFR Part 100#m as it relates to fission product releases for postulated accidents,fuel rod failure criteria should be given for all known fuel rod failure mechanisms. Fuel rod failure is defined as the loss of fuel rod hermeticity. Although we recognize that it is not possible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods, it is the objective of the review to assure that fuel does not fail due to specific causes during normal operation and anticipated operational occurrences. Fuel rod failures are permitted during postulated accidents, but they must be accounted for in the dose analysis.

Fuel rod failures can be caused by overheating, pellet/cladding interaction (PCI), hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. Fuel failure criteria should address the following to be complete.

(a) Hydriding: Hydriding as a cause of failure (i.e., primary hydriding) is prevented by keeping the level of moisture and other hydrogenous impurities very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 ppm. Current ASTM specifications (Ref. 7) for UO2 fuel pellets state an equivalent limit of 2 ppm of hydrogen from all sources. For other materials clad in Zircaloy tubing, an equivalent quantity of moisture of hydrogen can be tolerated. A moisture of 2 mg H2O per cm3 of hot void volume within the Zircaloy cladding has been shown (Ref. 8) to be insufficient for primary hydride formation.

(b) Cladding Collapse: If axial gaps in the fuel pellet column occur due to densification, the cladding has the potential of collapsing into a gap (i.e., flattening). Because of the large local strains that accompany this process, collapse (flattened) cladding is assumed to fail.

(c) Fretting: Fretting is a potential cause of fuel failure, but it is a gradual process that would not be effective during the brief duration of an abnormal operational occurrence of a postulated accident. Therefore, the fretting wear requirement in paragraph (c) of subsection II.A.1, Fuel Damage, is sufficient to preclude fuel failures caused by fretting during transients.

(d) Overheating of Cladding: It has been traditional practice to assume that failures will not occur if the thermal margin criteria (DNBR for PWRs and CPR for BWRs) are satisfied. The review of these criteria is detailed in SRP Section 4.4. For postulated accidents, the total number of fuel rods that exceed the criteria has been assumed to fail for radiological dose calculation purposes.

Although a thermal margin criterion is sufficient to demonstrate the avoidance of overheating from a deficient cooling mechanism, it is not a necessary condition (i.e., DNB is not a failure mechanism) and other mechanistic methods may be acceptable. There is at present little experience with other approaches, but new positions recommending different criteria should address cladding temperature, pressure, time duration, oxidation, and embrittlement.

(e) Overheating of Fuel Pellets: It has also been traditional practice to assume that failure will occur if centerline melting takes place. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. For normal operation and anticipated operational occurrences, centerline melting is not permitted. For postulated accidents, the total number of rods that experience centerline melting should be assumed to fail for radiological dose calculation purposes. The centerline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to come into contact with the cladding nor produce local hot spots. The assumption that centerline melting results in fuel failure is conservative.

(f) Excessive Fuel Enthalpy: For a severe reactivity initiated accident (RIA) in a BWR at zero or low power, fuel failure is assumed to occur if the radially averaged ruel rod enthalpy is greater than 170n cal/g at any axial location. For full-power RIAs in a BWR and all RIAs an a PWR, the thermal margin criteria (DNBR and CPR) are used as fuel failure criteria to meet the guidelines of Regulatory Guide 1.77 (Ref. 6) as it relates to fuel rod failure. The 170 cal/g enthalpy criterion is primarily intended to address cladding overheating effects, but it also indirectly addresses pellet/cladding interactions (PCI). Other criteria may be more appropriate for an RIA, but continued approval of this enthalpy criterion and the thermal margin criteria may be given until generic studies yield improvements.

(g) Pellet/Cladding Interaction: There is no current criterion for fuel failure resulting from PCI, and the design basis can only be stated generally. Two related criteria should be applied, but they are not sufficient to preclude PCI failures. (1) The uniform strain of the cladding should not exceed 1%. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gage lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Although observing this strain limit may preclude some PCI failures, it will not preclude the corrosion-assisted failures that occur at low strains, nor will it preclude highly localized overstrain failures. (2) Fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the cladding. Such a PCI is avoided by avoiding fuel melting. Note that this same criterion was invoked in paragraph (e) to ensure tht overheating of the cladding would not occur.

(h) Bursting: to meet the requirements of #mAppendix K#m of 10 CFR Part 50 (Ref. 9) as it relates to the incidence of rupture during a LOCA, a rupture temperature correlation must be used in the LOCA ECCS analysis. Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure. Although fuel suppliers may use different rupture-temperature vs differential-pressure curves, an acceptable curve should be similar to the one described in Ref. 10.

(i) Mechanical Fracturing: A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. Cladding integrity may be assumed if the applied stress is less than 90% of the irradiated yield stress at the appropriate temperature. Other proposed limits must be justified. Results from the seismic and LOCA analysis (see Appendix A to this SRP section) may show that failures by this mechanism will not occur for less severe events.

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(please press Space to continue)').

topic '10 CFR Part 100'.
text is read ('p100.ws').
write (con:,?text).
do ('I.B.1.b Fuel Rod Failure').
end.

topic 'Appendix K'.
text is read ('app_k.ws').
write (con:,?text).
do ('I.B.1.b Fuel Rod Failure').
end.

end.

topic 'I.B.1.c Fuel Coolability'.
say ('
4.2.II.A.3 Fuel Coolability
```

This subsection applies to postulated accidents, and most of the information to be reviewed will be contained in Chapter 15 of the Safety Analysis Report. Paragraph (e) addressed the combined effects of two accidents, however, and that information should be contained in Section 4.2 of the Safety Analysis Report. To meet the requirements of #mGeneral Design Criteria 27 and 35#m as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria should be given for all severe damage mechanisms. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. Control rod insertability criteria are also addressed in this subsection.

Such criteria should address the following to be complete:

(a) Cladding Embrittlement: To meet the requirements of 10 CFR Part 50, #m&50.46#m, as it relates to cladding embrittlement for a LOCA, acceptance criteria of 2200F on peak cladding temperature and 17% on maximum cladding oxidation must be met. (Note: If the cladding were predicted to collapse in a given cycle, it would also be predicted to fail and, therefore, should not be irradiated in that cycle; consequently, the lower peak cladding temperature limit of 1800F previously described in Ref. 11 is no longer needed.) Similar temperature and oxidation criteria may be justified for other accidents.

(b) Violent Expulsion of Fuel: In severe reactivity initiated accidents, such as rod ejection in a PWR or rod drop in a BWR, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and the rod-bundle geometry of the fuel and to produce pressure pulses in the primary system. To meet the guidelines of Regulatory Guide 1.77 as it relates to preventing widespread fragmentation and dispersal of the fuel and avoiding the generation of pressure pulses in the primary system of a PWR, a radially averaged enthalpy limit of 280 cal/g should be observed. This 280 cal/g limit should also be used for BWRs.

(c) Generalized Cladding Melting: Generalized (i.e., non-local) melting of the cladding could result in the loss of rod-bundle fuel geometry. Criteria for cladding embrittlement in paragraph (a) above are more stringent than melting criteria would be; therefore, additional specific criteria are not used.

(d) Fuel Rod Ballooning: To meet the requirements of #mAppendix K#m of 10 CFR Part 50 as it relates to degree of swelling, burst strain and flow blockage resulting from cladding ballooning (swelling) must be taken into account in the analysis of core flow distribution. Burst strain and flow blockage models must be based on applicable data (such as Refs. 10, 12, and 13) in such a way that (1) the temperature and differential pressure at which the cladding will rupture are properly estimated (see paragraph (h) of subsection II.A.2), (2) the resultant degree of cladding swelling is not underestimated, and (3) the associated reduction in assembly flow area is not underestimated. The flow blockage model evaluation is provided to the Reactor Systems Branch for incorporation in the comprehensive ECCS evaluation model to show that the 2200F cladding temperature and 17% cladding oxidation limits are not exceeded. The reviwer should also determine if fuel rod ballooning should be included in the analysis of other accidents involving system depressurization.

(e) Structural Deformation: Analytical procedures are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied

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    Forces".

(please press Space to continue)').

topic '&50.46'.
text is read ('p50_46.ws').
write (con:,?text).
do ('I.B.1.c Fuel Coolability').
end.

topic 'Appendix K'.
text is read ('app_k.ws').
write (con:,?text).
do ('I.B.1.c Fuel Coolability').
end.

end.

topic 'I.B.2 Description and Design Drawings'.
say ('
4.2.II.B Description and Design Drawings

The reviewer should see that the fuel system description and design drawings
are complete enough to provide an accurate representation and to supply infor-
mation needed in audit evaluations. Completeness is a matter of judgment, but
the following fuel system information and associated tolerances are necessary
for an acceptable fuel system description:

       Type and metallurgical state of the cladding
       Cladding outside diameter
       Cladding inside diameter
       Cladding inside roughness
       Pellet outside diameter
       Pellet roughness
       Pellet density
       Pellet resintering data
       Pellet length
       Pellet dish dimensions
       Burnable poison content
       Insulator pellet parameters
       Fuel column length
       Overall rod length
       Rod internal void volume
       Fill gas type and pressure
       Sorbed gas composition and content
       Spring and plug dimensions
       Fissile enrichment
       Equivalent hydraulic diameter
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        Coolant pressure

The following design drawing have also been found necessary for an acceptable
fuel system description:

        Fuel assembly cross section
        Fuel assembly outline
        Fuel rod schematic
        Spacer grid cross section
        Guide tuve and nozzle joint
        Control rod assembly cross section
        Control rod assembly outline
        Control rod schematic
        Burnable poison rod assembly cross section
        Burnable poison rod assembly outline
        Burnable poison rod schematic
        Orifice and source assembly outline

(please press Space to continue)').
end.

topic 'I.B.3 Design Evaluation'.
say ('
4.2.II.C Design Evaluation

The methods of demonstrating that the design bases are met must be reviewed.
Those methods include operating experience, prototype testing, and analytical
predictions. Many of these methods will be presented generically in topical
reports and will be incorporated in the Safety Analysis Report by reference.

(please press Space to continue)').
end.

topic 'I.B.3.a Operating Experience'.
say ('
4.2.II.C.1 Operating Experience

Operating experience with fuel systems of the same or similar design should
be described. When adherence to specific design criteria can be conclusively
demonstrated with operating experience, prototype testing and design analyses
that were performed prior to gaining that experience need not be reviewed.
Design criteria for fretting wear, oxidation, hydriding, and crud buildup
might be addressed in this manner.

(please press Space to continue)').
end.

topic 'I.B.3.b Prototype Testing'.
```

```
say ('
4.2.II.C.2 Prototype Testing

When conclusive operating experience is not available, as with the introduc-
tion of a design change, prototype testing should be reviewed. Out-of-reactor
tests should be performed when practical to determine the characteristics of
the new design. No definitive requirements have been developed regarding those
design features that must be tested prior to irradiation, but the following
out-of-reactor tests have been performed for this purpose and will serve as a
guide to the reviewer:

       Spacer grid structural tests
       Control rod structural and performance tests
       Fuel assembly structural tests (lateral, axial and torsional stiffness,
            frequency, and damping)
       Fuel assembly hydraulic flow tests (lift forces, control rod wear,
            vibration, and assembly wear and life)

In-reactor testing of design features and lead-assembly irradiation of whole
assemblies of a new design should be reviewed. The following phenomena that
have been tested in this manner in new designs will serve as a guide to the
reviewer:

       Fuel and burnable poison rod growth
       Fuel rod bowing
       Fuel assembly growth
       Fuel assembly bowing
       Channel box wear and distortion
       Fuel rod ridging (PCI)
       Crud formation
       Fuel rod integrity
       Holddown spring relaxation
       Spacer grid spring relaxation
       Guide tube wear characteristics

In some cases, in-reactor testing of a new fuel assembly design or a new
design feature cannot be accomplished prior to operation of a full core of
that design. This inability to perform in-reactor testing may result from an
incompatability of the new design with the previous design. In such cases,
special attention should be given to the surveillance plans (see subsection
II.D)

(please press Space to continue)').
end.

topic 'I.B.3.c Analytical Predictions'.
say ('
4.2.II.C.3 Analytical Predictions
```

Some design bases and related parameters can only be evaluated with calculational procedures. The analytical methods that are used to make performance predictions must be reviewed. Many such reviews have been performed establishing numerous examples for the reviewer. The following paragraph discuss the more established review patterns and provide many related references.

(a) Fuel Temperatures (Stored Energy): Fuel temperatures and stored energy during normal operation are needed as input to ECCS performance calculations. The temperature calculations require complex computer codes that model many different phenomena. Phenomenological models that should be reviewed include the following:

Radial power distribution
Fuel and cladding temperature distribution
Burnup distribution in the fuel
Thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers
Densification of the fuel
Thermal expansion of the fuel and cladding
Fission gas production and release
Solid and gaseous fission product swelling
Fuel restructuring and relocation
Fuel and cladding dimensional changes
Fuel-to-cladding heat transfer coefficient
Thermal conductivity of the gas mixture
Thermal conductivity in the Knudsen domain
Fuel-to-cladding contact pressure
Heat capacity of the fuel and cladding
Growth and creep of the cladding
Rod internal gas pressure and composition
Sorption of helium and other fill gases
Cladding oxide and crud layer thickness
Cladding-to-coolant heat transfer coefficient

Because of the strong interaction between these models, overall code behavior must be checked against data (standard problems or benchmarks) and the NRC audit codes (Refs. 14 and 15). Examples of previous fuel performance code reviews are given in References 16 through 20.

(b) Densification Effects: In addition to its effect on fuel temperatures (discussed above), densification affects (1) core power distributions (power spiking, see #mSRP Section 4.3#m), (2) the fuel linear heat generation rate (LHGR, see #mSRP Section 4.4#m), and (3) the potential for cladding collapse. Densification magnitudes for power spike and LHGR analyses are discussed in References 21 and in Regulatory Guide 1.126 (Ref. 22). To be acceptable, densification models should follow the guidelines of Regulatory Guide 1.126. Models for cladding-collapse times must also be reviewed, and previous review examples are given in References 23 and 24.

(c) Fuel Rod Bowing: Guidance for the analysis of fuel rod bowing is given in References 25. Interim methods that may be used prior to compliance with this guidance are given in Reference 26. At this writing, the causes of fuel rod bowing are not well understood and mechanistic analyses of rod bowing are not being approved.

(d) Structural Deformation: Acceptance Criteria are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

(e) Rupture and Flow Blockage (Ballooning): Zircaloy rupture and flow blockage models are part of the ECCS evaluation model and should be reviewed by CPB. The models are empirical and should be compared with relevant data. Examples of such data and previous reviews are contained in References 10, 12, and 13.

(f) Fuel Rod Pressure: The thermal performance code for calculating temperatures discussed in paragraph (a) above should be used to calculate fuel rod pressures in conformance with fuel damage criteria of Subsection II.A.1, paragraph (f). The reviewer should ensure that conservatisms that were incorporated for calculating temperatures do not introduce nonconservatisms with regard to fuel rod pressures.

(g) Metal/Water Reaction Rate: To meet the requirements of #mAppendix K#m of 10 CFR Part 50 (Ref. 9) as it relates to metal/water reaction rate, the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction should be calculated using the Baker-Just equation (Ref. 27). For non-LOCA applications, other correlations may be used if justified.

(h) Fission Product Inventory: To meet the guidelines of Regulatory Guides 1.3, 1.4, 1.25 and 1.77 (Refs. 6, 28-30) as they relate to fission pro duct release, the available radioactive fission product inventory in fuel rods (i.e., the gap inventory) is presently specified by the assumptions in those Regulatory Guides. These assumptions should be used until improved calculational methods are approved by CPB (see Ref. 31).

(please press Space to continue)').

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topic 'Appendix K'.
text is read ('app_k.ws').
write (con:,?text).
do ('I.B.3.c Analytical Predictions').
end.

end.
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```
topic 'I.B.4 Testing, Inspection, and Surveillance Plans'.
say ('
4.2.II.D Testing, Inspection, and Surveillance Plans

Plans must be reviewed for each plant for testing and inspection of new fuel
and for monitoring and surveillance of irradiated fuel.

(please press Space to continue)').
end.

topic 'I.B.4.a Testing and Inspection of New Fuel'.
say ('
4.2.II.D.1 Testing and Inspection of New Fuel

Testing and inspection plans for new fuel should include verification of clad-
ding integrity, fuel system dimensions, fuel enrichment, burnable poison con-
centration, and absorber composition. Details of the manufacturer''s testing
and inspection programs should be documented in quality control reports, which
should referenced and summarized in the Safety Analysis Report. The program
for onsite inspection of new fuel and control assemblies after they have been
delivered to the plant should also be described. Where the overall testing and
inspection programs are essentially the same as for previously approved plants,
a statement to that effect should be made. In that case, the details of the
programs need not be included in the Safety Analysis Report, but an appropri-
ate reference should be cited and a (tabular) summary should be presented.

(please press Space to continue)').
end.

topic 'I.B.4.b On-line Fuel System Monitoring'.
say ('
4.2.II.D.2 On-line Fuel System Monitoring

The applicant''s on-line fuel rod failure detection methods should be revi-
ewed. Both the sensitivity of the instruments and the applicant''s commitment
to use the instruments should be evaluated. References 32 and 33 evaluate
several common detection methods and should be utilized in this review.

Surveillance is also needed to assure that B4C control rods are not losing
reactivity. Boron compounds are susceptible to leaching in the event of a
cladding defect. Periodic reactivity worth tests such as described in Refer-
ence 34 are acceptable.

(please press Space to continue)').
end.

topic 'I.B.4.c Post-irradiation Surveillance'.
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say ('
4.2.II.D.3 Post-irradiation Surveillance

A post-irradiation fuel surveillance program should be described for each
plant to detect anomalies or confirm expected fuel performance. The extent of
an acceptable program will depend on the history of the fuel design being con-
sidered, i.e., whether the proposed fuel design is the same as current opera-
ting fuel or incorporates new design features.

For a fuel design like that in other operating plants, a minimum acceptable
program should include a qualitative visual examination of some discharged
fuel assemblies from each refueling. Such a program should be sufficient to
identify gross problems of structural integrity, fuel rod failure, rod bowing,
or crud deposition. There should also be a commitment in the program to
perform additional surveillance if unusual behavior is noticed in the visual
examination or if plant instrumentation indicates gross fuel failures. The
surveillance program should address the disposition of failed fuel.

In addition to the plant-specific surveillance program, there should exists a
continuing fuel surveillance effort for a given type, make, or class of fuel
that can be suitably referenced by all plants using similar fuel. If the ab-
sence of such a generic program, the reviewer should expect more detail in
the plant-specific program.

For a fuel design that introduces new features, a more detailed surveillance
program commensurate with the nature of the changes should be described. This
program should include appropriate qualitative and quantitative inspections to
be carried out at interim and end-of-life refueling outages. This surveillance
program should be coordinated with prototype testing discussed in subsection
II.C.2. When prototype testing cannot be performed, a special detailed surve-
illance program should be planned for the first irradiation of a new design.

(please press Space to continue)').
end.

topic 'I.C Review Procedures'.
say ('
4.2.III Review Procedures

For construction permit (CP) applications, the review should assure that the
design bases set forth in the Preliminary Safety Analysis Report (PSAR) meet
the acceptance criteria given in #mSubsection II.A#m. The CP review should
further determine from a study of the preliminary fuel system design that
there is reasonable assurance that the final fuel system design will meet the
design bases. This judgment may be based on experience with similar designs.

For operating license (OL) applications, the review should confirm that the
design bases set forth in the Final Safety Analysis Report (FSAR) meet the
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acceptance criteria given in Subsection II.A and that the final fuel system design meets the design bases.

Much of the fuel system review is generic and is not repeated for each similar plant. That is, the reviewer will have reviewed the fuel design or certain aspects of the fuel design in previous PSARs, FSARs, and licensing topical reports. All previous reviews on which the current review is dependent should be referenced so that a completely documented safety evaluation is contained in the plant safety evaluation report. In particular, the NRC safety evaluation reports for all relevant licensing topical reports should be cited. Certain generic reviews have also been performed by CPB reviewers with findings issued as NUREG- or WASH-series reports. At the present time these reports include References 9, 11, 21, 31, 32, 35, and 36, and they should all be appropriately cited in the plant safety evaluation report. Applicable Regulatory Guides (Refs. 6, 22, 28-30, and 41) should also be mentioned in the plant safety evaluation reports. Deviation from these guides or positions should be explained. After briefly discussing related previous reviews, the plant safety evaluation should concentrate on areas where the applications is not identical to previously reviewed and approved applications and areas related to newly discovered problems.

Analytical predictions discussed in #mSubsection II.C.3#m will be reviewed in RSARs, FSARs, or licensing topical reports. When the methods are being reviewed, calculations by the staff may be performed to verify the adequacy of the analytical methods. Thereafter, audit calculations will not usually be performed to check the results of an approved method that has been submitted in a Safety Analysis Report. Calculations, benchmarking exercises, and additional reviews of generic methods may be undertaken, however, at any time the clear need arises to reconfirm the adequacy of the method.

(please press Space to continue)').

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topic 'Subsection II.A'.
say ('



This is part 4.2.II.A Design Bases
(please press Space to continue)')
and do ('I.C Review Procedures').
end.

topic 'Subsection II.C.3'.
say ('



This is part 4.2.II.C.3 Analytical Predictions
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(please press Space to continue)')
and do ('I.C Review Procedures').
end.

end.

topic 'I.D Evaluation Findings'.
say ('
4.2.IV Evaluation Findings

The reviewer should verify that sufficient information has been provided to
satisfy the requirements of this SRP section and that the evaluation supports
conclusions of the following type, to be included in the staff''s safety
evaluation report:

The staff concludes that the fuel system of the ____________ plant has been
designed so that (a) the fuel system will not be damaged as a result of normal
operation and anticipated operational occurrences, (b) fuel damage during pos-
tulated accidents would not be severe enough to prevent control rod insertion
when it is required, and (c) core coolability will always be maintained, even
after severe postulated accidents and thereby meets the related requirements
of 10 CFR Part 50, &50.46; 10 CFR Part 50, Appendix A, General Design Criteria
10, 27, and 35; 10 CFR Part 50, Appendix K; and 10 CFR Part 100. This conclu-
sion is based on the following:

        1. The applicant has provided sufficient evidence that these design
           objectives will be met based on operating experience, prototype
           testing, and analytical predictions. Those analytical predictions
           dealing with structural response, control rod ejection (PWR) or drop
           (BWR), and fuel densification have been performed in accordance with
           (a) the guidelines of Regulatory Guides 1.60, 1.77, and 1.126, or
           methods that the staff has reviewed and found to be acceptable
           alternatives to those Regulatory Guides, and (b) the guidelines for
           "Evaluation of Fuel Assembly Structural Response to Externally
           Applied Forces" in Appendix A to SRP Section 4.2.

        2. The applicant has provided for testing and inspection of new fuel to
           insure that it is within design tolerances at the time of core
           loading. The applicant has made a commitment to perform on-line fuel
           failure monitoring and postirradiation surveillance to detect ano-
           malies or confirm that the fuel has performed as expected.

The staff concludes that the applicant has described methods of adequately
predicting fuel rod failures during postulated accidents so that radioactivity
releases are not underestimated and thereby meets the related requirements of
10 CFR Part 100. In meeting these requirements, the applicant has (a) used the
fission-product release assumptions of Regulatory Guides 1.3 (or 1.4), 1.25,
and 1.77 and (b) performed the analysis for fuel rod failures for the rod
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ejection accident in accordance with the guidelines of Regulatory Guide 1.77
or with methods that the staff has reviewed and found to be an acceptable
alternative to Regulatory Guide 1.77.

(please press Space to continue)').
end.

topic 'I.E Implementation'.
say ('
4.2.V Implementation

The following is intended to provide guidance to applicants and licensees
regarding the NRC staff''s plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alterna-
tive method for complying with specified portions of the Commission''s regu-
lations, the method described herein will be used by the staff in its evalu-
ation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed
herein are contained in the referenced regulatory guides and NUREGs.

(please press Space to continue)').
end.

topic 'I.F References'.
say ('
4.2.VI References

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end.

topic 'I.G Appendix'.
say ('
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4.2.VII Appendix

EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE
TO EXTERNALLY APPLIED FORCES
TO
STANDARD REVIEW PLAN SECTION 4.2

A. BACKGROUND

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. This Appendix describes the review that should be performed of the fuel assembly structural response to seismic and LOCA loads. Background material for this Appendix is given in References 37-40.

B. ANALYSIS OF LOADS

1. Input

Input for the fuel assembly structural analysis comes from results of the primary coolant system and reactor internals structural analysis, which is reviewed by the Mechanical Engineering Branch. Input for the fuel assembly response to a LOCA should include (a) motions of the cor plate, core shroud, fuel alignment plate, or other relevant structures; these motions should correspond to the break that produced the peak fuel assembly loadings in the primary coolant system and reactor internals analysis, and (b) transient pressure differences that apply loads directly to the fuel assembly. If the earthquake loads are large enough to produce a non-linear fuel assembly response, input for the seismic analysis should use structure motions corresponding to the reactor primary coolant system analysis for the SSE; if a linear response is produced, a spectral analysis may be used in accordance with the guidelines of Regulatory Guide 1.60 (Ref. 41).

2. Methods

Analytical methods used in performing structural response analyses should be reviewed. Justification should be supplied to show that the numerical solution techniques are appropriate.

Linear and non-linear structural representations (i.e., the modeling) should also be reviewed. Experimental verification of the analytical representation of the fuel assembly components should be provided when practical

A sample problem of a simplified nature should be worked by the applicant and compared by the reviewer with either hand calculations or results generated by the reviewer with an independent code (Ref.38). Although the sample problem should use a structural representation that is as close as possible to the design in question (and, therefore, would vary from one vendor to another), simplifying assumptions may be made (e.g., one might use a 3-assembly core region with continuous sinusoidal input).

The sample problem should be designed to exercise various features of the code and reveal their behavior. The sample problem comparison is not, however, designed to show that one code is more conservative than another, but rather to alert the reviewer to major discrepancies so that an explanation can be sought.

3. Uncertainty Allowances

The fuel assembly structural models and analytical methods are probably conservative and input parameters are also conservative. However, to ensure that the fuel assembly analysis does not introduce any non-conservatisms, two precautions should be taken: (a) If it is not explicitly evaluated, impact loads from the PWR LOCA analysis should be increased (by about 30%) to account for a pressure pulse, which is associated with steam flashing that affects only the PWR fuel assembly analysis. (b) Conservative margin should be added if any part of the analysis (PWR or BWR) exhibits pronounced sensitivity to input variations.

Variations in resultant loads should be determined for +-10% variations in input amplitude and frequency; variations in amplitude and frequency should be made separately, not simultaneously. A factor should be developed for resultant load magnitude variations of more than 15%. For example, if +-10% variations in input magnitude or frequency produce a maximum resultant increase of 35%, the sensitivity factor would be 1.2. Since resonances and pronounced sensitivities may be plant-dependent, the sensitivity analysis should be performed on a plant-by-plant basis until the reviewer is confident that further sensitivity analyses are unnecessary or it is otherwise demonstrated that the analyses performed are bounding.

4. Audit

Independent audit calculations for a typical full-sized core should be performed by the reviewer to verify that the overall structural representation is adequate. An independent audit code (Ref. 38) should be used for this audit during the generic review of the analytical methods.

5. Combination of loads

To meet the requirements of #mGeneral Design 2#m as it relates to combining loads, an appropriate combination of loads from natural phenomena and accident conditions must be made. Loads on fuel assembly component should be calculated for each input (i.e., seismic and LOCA) as described above in Paragraph 1, and the resulting loads should be added by the square-root-of-sum-of-squares (SRSS) method. These combined loads should be compared with the component strengths described in Section C according to the acceptance criteria in Section D.

C. DETERMINATION OF STRENGTH

1. Grids

All modes of loading (e.g., in-grid and through-grid loadings) should be considered, and the most damaging mode should be represented in the vendor''s laboratory grid strength tests. Test procedures and results should be reviewed to assure that the appropriate failure mode is being predicted. The review should also confirm that (a) the testing impact velocities correspond to expected fuel assembly velocities, and (b) the crushing load P(crit) has been suitably selected from the load-vs-deflection curves. Because of the potential for different test rigs to introduce measurement variations, an evaluation of the grid strength test equipment will be included as part of the review of the test procedure.

The consequences of grid deformation are small. Gross deformation of grids in many PWR assemblies would be needed to interfere with control rod insertion during an SSE (i.e., buckling of a few isolated grids could not displace guide tubes significantly from their proper location), and grid deformation (without channel deflection) would not affect control blade insertion in a BWR. In a LOCA, gross deformation of the hot channel in either a PWR or a BWR would result in only small increases in peak cladding temperature. Therefore, average values are appropriate, and the allowable crushing load P(crit) should be the 95% confidence level on the true mean as taken from the distribution of measurements on unirradiated production grids at (or corrected to) operating temperature. While P(crit) will increase with irradiation, ductility will be reduced. The extra margin in P(crit) for irradiated grids is thus assumed to offset the unknown deformation behavior of irradiated grids beyond P(crit).

2. Components Other than Grids

Strength of fuel assembly components other than spacer grids may be deduced from fundamental material properties or experimentation. Supporting evidence for strength values should be supplied. Since structural failure of these components (e.g., fracturing of guide tubes or fragmentation of fuel rods) could be more serious than grid deformation, allowable values should bound a large percentage (about 95%) of the distribution of component strengths. Therefore, ASME Boiler and Pressure Vessel Code values and procedures may be used where appropriate for determining yield and ultimate strengths. Specification of allowable values may follow the ASME Code requirements and should include consideration of buckling and fatigue effects.

D. ACCEPTANCE CRITERIA

1. Loss-of-Coolant Accident

Two principal criteria apply for the LOCA: (a) fuel rod fragmentation must not occur as a direct result of the blowdown loads, and (b) the 10 CFR Part 50, &50.46 temperature and oxidation limits must not be exceeded. The first criterion is satisfied if the combined loads on the fuel rods and components other than grids remain below the allowable values defined above. The second criterion is satisfied by an ECCS analysis. If combined loads on the grids remain below P(crit), as defined above, then no significant distortion of the fuel assembly would occur and the usual ECCS analysis is sufficient. If combined grid loads exceed P(crit), then grid deformation must be assumed and the ECCS analysis must include the effects of distorted fuel assemblies. An assumption of maximum credible deformation (i.e., fully collapsed grids) may be made unless other assumptions are justified.

Control rod insertability is a third criterion that must be satisfied.

Loads from the worst-case LOCA that requires control rod insertion must be combined with the SSE loads, and control rod insertability must be demonstrated for that combined load. For a PWR, if combined loads on the grids remain below P(crit) as defined above, then significant deformation of the fuel assembly would not occur and control rod insertion would not be interfered with by lateral displacement of the guide tubes. If combined loads on the grids exceed P(crit), then additional analysis is needed to show that deformation is not severe enough to prevent control rod insertion.

For a BWR, several conditions must be met to demonstrate control blade insertability: (a) combined loads on the channel box must remain below the allowable value defined above for components other that grids; otherwise, additional analysis is needed to show that deformation is not severe enough to prevent control blade insertion, and (b) vertical liftoff forces must not unseat the lower tieplate from the fuel support piece such that the resulting loss of lateral fuel bundle positioning could interfere with control blade insertion.

2. Safe Shutdown Earthquake

Two criteria apply for the SSE: (a) fuel rod fragmentation must not occur as a result of the seismic loads, and (b) control rod insertability must be assured. The first criterion is satisfied by the criteria in Paragraph 1. The second criterion must be satisfied for SSE loads alone if no analysis for combined loads is required by Paragraph 1.

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topic 'II Nuclear Design'.
ask ('
4.3 NUCLEAR DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)
Secondary - None

This part contains:', part,
[
'4.3.I Areas of Review',
'4.3.II Acceptance Criteria',
'4.3.III Review Procedures',
'4.3 IV Evaluation Finding',
'4.3 V Implementation',
'4.3 VI References',
'4.3 VII Appendix',
'Content',
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'Index',
'Opening Menu']).

do (?part).
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topic 'II.A Areas of Review'.
say ('
4.3.I Areas of Review

The review of the nuclear design of the fuel assemblies, control systems, and
reactor core is carried out to aid in confirming that fuel design limits will
not be exceeded during normal operation or anticipated operational transients,
and that the effects of postulated reactivity accidents will not cause signi-
ficant damage to the reactor coolant pressure boundary or impair the capabi-
lity to cool the core and to assure conformance with the requirements of
#mGeneral Design Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28#m.

The review of the nuclear design under this SRP section, the review of the
fuel system design under #mSRP Section 4.2#m, the review of the thermal and
hydraulic design under #mSRP Section 4.4#m, and the review of the transient
and accident analyses under the SRP section for Chapter 15 of the applicant''s
safety analysis report (SAR), are all necessary in order to confirm that the
requirements defined above are met.

The specific areas of interest in the nuclear design include:

1. Confirmation that design bases are established as required by the appro-
   priate General Design Criteria.

2. The areas concerning core power distribution. These are:

   a. The presentation of expected or possible power distributions including
      normal and extreme cases for steady-state and allowed load-follow tran-
      sients and covering a full range of reactor conditions of time in cycle,
      allowed control rod positions, and possible fuel burnup distributions.

   b. The presentation of the core power distributions as axial, radial, and
      local distributions and peaking factors to be used in the transient and
      accident analyses. The effects of phenomena such as fuel densification
      should be included in these distributions and factors.

   c. The translation of the design power distributions into operating power
      distributions, including instrument-calculation correlations, operating
      procedures and measurements, and necessary limits on these operations.

   d. The requirements for instruments, the calibration and calculations
      involved in their use, and the uncertainties involved in translation of
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instrument readings into power distributions.

e. Limits and setpoints for actions, alarms, or scram for the instrument systems and demonstration that these systems can maintain the reactor within design power distribution limits.

f. Measurements in previous reactors and critical experiments and their use in the uncertainty analyses, and measurements to be made on the reactor under review, including startup confirmatory tests and periodically required measurements.

g. The translation of design limits, uncertainties, operating limits, instrument requirements, and setpoints into technical specifications.

3. The areas concerning reactivity coeffients. These are:

a. The applicant''s presentation of calculated nominal values for the reactivity coefficients such as the moderator coefficient, which involves primarily effects from density changes and takes the form of temperature, void, or density coefficients; the Doppler coefficient; and power coefficients. The range of reactor states to be covered includes the entire operating range from cold shutdown through full power, and the extremes reached in transient and accident analyses. It includes the extremes of time in cycle and an appropriate range of control rod insertions for the reactor states.

b. The applicant''s presentation of uncertainty analyses for nominal values including the magnitude of the uncertainty and the justification of the magnitude by examination of the accuracy of the methods used in calculations (SAR Section 4.3.3), and comparison where possible with reactor experiments.

c. The applicant''s combination of nominal values and uncertainties to provide suitably conservative values for use in reactor steady-state analysis (primarily control requirements, SAR Section 4.3.2.4), stability analyses (SAR Section 4.3.2.8), and the transient and accident analyses presented in SAR Chapter 15.

4. The areas concerning reactivity control requirements and control provisions. These are:

a. The control requirements and provisions for control necessary to compensate for long-term reactivity changes of the core. These reactivity changes occur because of depletion of the fissile material in the fuel, depletion of burnable poison in some of the fuel rods, and buildup of fission products and transuranium isotopes.

b. The control requirements and provisions for control needed to compensate for the reactivity change caused by changing the temperature of the reactor from the hot, zero power condition to the cold shutdown condition.

c. The control requirements and provisions for control needed to compensate for the reactivity effects caused by changing the reactor power level from full power to zero power.

d. The control requirements and provisions for control needed to compensate for the effects on the power distribution of the high cross-section Xe-135 isotope.

e. The adequacy of the control systems to assure that the reactor can be returned to and maintained in the cold shutdown condition at any time during operation.

f. The applicant''s analysis and experimental basis for determining the reactivity worth of a "stuck" control rod of highest worth.

g. The provision of two independent control systems.

5. The areas of control rod patterns and reactivity worths. These are:

   a. Descriptions and figures indicating the control rod pattern expected to be used throughout a fuel cycle. This includes operation of single rods or of groups or banks of rods, rod withdrawal order, and insertion limits as a function of power and core life.

   b. Descriptions of allowable deviations from the patterns indicated above, such as for misaligned rods, stuck rods, or rod positions used for spatial power shaping.

   c. Descriptions, tables, and figures of the maximum worths of individual rods or banks as a function of position for power and cycle life conditions appropriate to rod withdrawal transients and rod ejection or drop accidents. Descriptions and curves of maximum rates of reactivity increase associated with rod withdrawals, experimental confirmation of rod worths or other factors justifying the reactivity increase rates used in control rod accident analyses, and equipment, administrative procedures, and alarms which may be employed to restrict potential rod worths should be included.

   d. Descriptions and graphs of scram reactivity as a function of time after scram initiation and other pertinent parameters, including methods for calculating the scram reactivity.

6. The area of criticality of fuel assemblies. Discussions and tables giving values of Keff for single assemblies and groups of adjacent fuel assemblies

up to the number required for criticality, assuming the assemblies are dry and also immersed in water, are reviewed.

7. The areas concerning analytical methods. These are:

   a. Description of the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients, burnup, and stability.

   b. The data base used for neutron cross-sections and other nuclear parameters.

   c. Verification of the analytical methods by comparison with measured data.

8. The areas concerning pressure vessel irradiation. These are:

   a. Neutron flux spectrum above 1 MeV in the core, at the core boundaries, and at the inside pressure vessel wall.

   b. Assumptions used in the calculations; these include the power level, the use factor, the type of fuel cycle considered, and the design life of the vessel.

   c. Computer codes used in the analysis.

   d. The data base for fast neutron cross sections.

   e. The geometric modeling of the reactor, support barrel, water annulus, and pressure vessel.

   f. Uncertainties in the calculation.

9. The adequacy of limits on power distribution during normal operation is reviewed in connection with review of the thermal-hydraulic design under #mSRP Section 4.4#m.

10 The adequacy of proposed instrumentation to meet the requirements for maintaining the reactor operating state within defined limits is reviewed under SRP Section 7.1 through 7.6.

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end.

topic 'II.B Acceptance Criteria'.
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4.3.II Acceptance Criteria

The acceptance criteria in the area of nuclear design are based on meeting the relevant requirements of the General Design Criteria (GDC) related to the reactor core and reactivity control system (Ref. 1). The relevant requirements are as follows:

A. GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.

B. GDC 11 requires that in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

C. GDC 12 requires that power oscillations which could result in conditions exceeding specified acceptable fuel design limits are nor possible or can be reliably and readily detected and suppressed.

D. GDC 13 requires provision of instrumentation and controls to monitor variables and systems that can affect the fission process over anticipated ranges for normal operation, anticipated operationa occurences and accident conditions, and to maintain the variables and systems within prescribed operating ranges.

E. GDC 20 requires automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety under accident conditions.

F. GDC 25 requires that no single malfunction of the reactivity control system (this does not include rod ejection or dropout) causes violation of the acceptable fuel design limits.

G. GDC 26 requires that two independent reactivity control systems of different design be provided, and that each system have the capability to control the rate of reactivity changes resulting from planned, normal power changes. One of the systems must be capable of reliably controlling anticipated operational occurrences. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.

H. GDC 27 requires that the reactivity control systems have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.

I. GDC 28 requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor cause sufficient damage to impair significantly the capability to cool the core.

The following discussions present less formal criteria and guidelines used in the review of the nuclear design for meeting the relevant requirements of the GDCs identified above.

1. There are no direct or explicit criteria for the power densities and power distributions allowed during (and at the limits of) normal operation, either steady-state or load-following. These limits are determined from an integrated consideration of fuel limits (SAR Section 4.2), thermal limit (SAR Section 4.4), scram limits (SAR Chapter 7) and transient and accident analyses (SAR Chapter 15). The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during anticipated transients and that other limits, such as the 2200F peak cladding temperature allowed for loss-of-coolant accidents (LOCA), are not exceeded during design basis accident. The limiting power distributions are then determined such that the limits on power densities and peaking factors can be maintained in operation. These limiting power distributions may be maintained (i.e., not exceeded) administratively (i.e., not by automatic scrams), provided a suitable demonstration is made that sufficient, properly translated information and alarms are available from the reactor instrumentation to keep the operator informed.

   The acceptance criteria in the area of power distribution are that the information presented should satisfactorily demonstrate that:

   a. A reasonable probability exists that the proposed design limits can be met within the expected operational range of the reactor, taking into account the analytical methods and data for the design calculations; uncertainty analyses and experimental comparisons presented for the design calculations; the sufficiency of design cases calculated covering times in cycle, rod positions, load-follow transients, etc.; and special problems such as power spikes due to densification, possible asymmetries, and misaligned rods.

   b. A reasonable probability exists that in normal operation the design limits will not be exceeded, based on consideration of information received from the power distribution monitoring instrumentation; the processing of that information, including calculations involved in the processing; the requirements for periodic check measurements; the accuracy of design calculations used in developing correlations when primary variables are not directly measured; the uncertainty analyses for the information alarms for the limits of normal operation (e.g., offset limits, control bank limits) and for abnormal situations (e.g., tilt alarms for control rod misalignment).

   Criteria for acceptable values and uses of uncertainties in operation, instrumentation numerical requirements, limit settings for alarms or scram, frequency and extent of power distribution measurements, and use of excore and incore instruments and related correlations and limits for offsets and tilts, all vary with reactor type. They can be found in staff safety evaluation reports and in appropriate sections of the technical specifications and accompanying bases for reactors similar to the reactor under review. The CPB has enunciated Branch Technical Position CPB 4.3-1 for Westinghouse reactors which employ constant axial offset control (Ref. 2).

Acceptance criteria for power spike models can be found in a staff technical report on fuel densification (Ref. 3).

Generally, special or newly emphasized problems related to core power distributions will not be a direct part of normal reviews but will be handled in special generic reviews. Fuel densification effects and the related power spiking and the use of uncertainties in design limits are examples of these areas.

2. The only directly applicable GDC in the area of reactivity coefficients is GDC 11, which states " ... the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity ", and is considered to be satisfied in light water reactors by the existence of the Doppler and negative power coefficients. There are no criteria that explicitly establish acceptable ranges of coefficient values or preclude the acceptability of a positive moderator temperature coefficient such as may exist in pressurized water reactors at beginning of core life.

   The acceptability of the coefficients in a particular case in determined in the reviews of the analyses in which they are used, e.g., control requirement analyses, stability analyses, and transient and accident analyses. The use of spatial effects such as weighting approximations as appropriate for individual transients are included in the analysis reviews. The judgment to be made under this SRP section is whether the reactivity coefficients have been assigned suitably conservative values by the applicant. The basis for that judgment includes the use to be made of a coefficient, i.e., the analyses in which it is important; the state of the art for the coefficient; the uncertainty associated with such calculations; experimental checks of the coefficient in operating reactors; and any required checks of the coefficient in the startup program of the reactor under review.

3. Acceptance criteria relative to control rod patterns and reactivity worths include:

   a. The predicted control rod worths and reactivity insertion rates must be reasonable bounds to values that may o#cur in the reactor. These values are used in the transient and accident analyses and judgment as to the adequacy of the uncertainty allowances are made in the review of the transient and accident analyses.

   b. Equipment, operating limits, and procedures necessary to restrict potential rod worths or reactivity insertion rates should be shown to be capable of performing these functions. It is a CPB position to require, where feasible, an alarm when any limit or restriction is violated or is about to be violated.

4. There are no specific criteria that must be met by the analytical methods or data that are used by an applicant or reactor vendor. In general, the analytical methods and data base should be representative of the state of the art, and the experiments used to validate the analytical methods should be adequate and encompass a sufficient range.

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topic 'II.C Review Procedures'.
say ('
4.3.III Review Procedures

The review procedures below apply in general to both the construction permit (CP) and operating licence (OL) stage reviews. At the CP stage, parameter values and certain design aspects may be preliminary and subject to change. At the OL stage, final values of parameters should be used in the analyses presented in the SAR. The review of the nuclear design of a plant is based on the information provided by the applicant in the safety analyses report, as amended, and in meetings and discussions with the applicant and his contractors and consultants. This review in some cases will be supplemented by independent calculations performed by the staff or staff consultants. Files of audit calculations are maintained by CPB for reference by the reviewer.

1. The reviewer confirms, as part of the review of specific areas of the nuclear design outlined below, that the design bases, design features, and design limits are established in conformance with the GDCs listed in subsection II of this SRP section.

2. The reviewer examines the information presented in the SAR to determine that the core power distributions for the reactor can reasonably be expected to fall within the design limits throughout all normal (steady-state and load-follow) operations, and that the instrument systems employed, along with the information processing systems and alarms, will reasonably assure that the maintenance of the distributions within these limits for normal operation.

   For a normal review, many areas related to core power distribution will have been examined in generic reviews or earlier reviews of reactors with generally similar core characteristics and instrument systems. A large part of the review on a particular case may then involve comparisons with information from previous application reviews. The comparisons may involve the shapes and peaking factors of normal and limiting distributions over the range of operating states of the reactor, the effects of power spikes from densification, assigned uncertainties and their use, calculation methods and data used, correlations used in control processes, instrumentation requirements, information processing methods including computer use, setpoints for operational limits and alarm limits, and alarm limits for abnormalities such as flux asymmetries.

An important part of this review, at the OL stage, covers the relevant sections of the proposed technical specifications, where power distributions and related controls such as control rod limits are discussed. Here the instrument requirements, limit settings, and measurement frequencies and requirements are set forth in full detail. The comparison of technical specifications should reveal any differences between essentially identical reactors or any lack of difference between reactors with changed core characteristics. Where these occur, the reviewer must assess the significance and validity of the differences or lack of differences.

This review and comparison may be supplemented with examinations of related topical reports from reactor vendors, generic studies by staff consultants, and startup reports from operating reactors which contain information on measured power distributions.

3. The reviewer determines from the applicant''s presentations that suitably conservative reactivity coefficients have been developed for use in reactor analyses such as those for control requirements, stability, and transients and acc)dents. The reviewer examines:

   a. The applicability and accuracy of methods used for calculations including the use of more accurate check calculations.

   b. The models involved in the calculations such as the model used for effective fuel temperature in Doppler coefficient analyses.

   c. The reactor state conditions assumed in determining values of the coefficients. For example, the pressurized water reactor (PWR) moderator temperature coefficient to be used in the steam lime break analysis is usually based on the reactor condition at end of cycle with all control rods inserted except the most reactive rod, and the moderator temperature in the hot standby range.

   d. The applicability and accuracy of experimental data from critical experiments and operating reactors used to determine or justify uncertainty allowances. Measurements during startup and during the cycle of moderator temperature coefficients and full power Doppler coefficients in the case of PWRs, and results of measurements of transients during startup in the case of boiling water reactors (BWRs), should be examined. As part of the review, comparisons are made between the values and uncertainty allowances for reactivity coefficients for the reactor under review and those for similar reactors previously reviewed and approved. Generally, many essential areas will have been covered during earlier reviews of similar reactors. The reviewer notes any differences in results for essentially identical reactors and any lack of differences for reactors with changed core characteristics, and judges the significance and validity of any differences or lack of differences.

4. The review procedures in the area of reactivity control requirements and control provisions are as follows:

   a. The reviewer determines that two independent reactivity control systems of different design are provided.

   b. The reviewer examines the tabulation of control requirements, the associated uncertainties, and the capability of the control systems, and determines by inspection and study of the analyses and experimental data that the values are realistic and conservative.

   c. The reviewer determines that one of the control systems is capable of returning the reactor to the cold shutdown condition and maintaining it in this condition, at any time in the cycle. It is necessary that proper allowance be made for all of the mechanisms that change the reactivity of the core as the reactor is taken from the cold shutdown state to the hot, full power operating state. The reviewer should determine that proper allowance is made for the decrease in fuel temperature, moderator temperature, and the loss of voids (in BWRs) as the reactor goes from the power operating range to cold shutdown.

   d. The reviewer determines that one of the control systems is capable of rapidly returning the reactor to the hot standby (shutdown) condition from any power level at any time in the cycle. This requirement is met by rapid insertion of control rods in all current light water reactors. Proper allowance for the highest worth control rod being stuck in the full-out position must be made. In PWRs, operational reactivity control is carried out by movement of control rods and by adjustments of the concentration of soluble poison in the coolant. The reviewer must pay particular attention to the proposed rod insertion limits in the power operating range, to assure that the control rods are capable of rapidly reducing the power and maintaining the reactor in the hot standby condition. This is an important point because the soluble poison concentration in the coolant could be decreased in order to raise reactor power, while the control rods were left inserted so far that in the event of a scram (rapid insertion of control rods), the available reactivity worth of the control rods on full insertion would not be enough to shut the reactor down to the hot standby condition.

   e. The reviewer determines that each of the independent reactivity control systems is capable of controlling the reactivity changes resulting from planned, normal power operation. This determination is made by comparing the rate of reactivity change resulting from planned, normal operation to the capabilities of each of the two control systems. Sufficient margin must exist to allow for the uncertainties in the rate.

5. The review procedures in the area of control rod patterns and reactivity worths are:

   a. The reviewer determines by inspection and study of the information described in subsection I.5 of this SRP section that the control rod and bank worts are reasonable. This determination involves evaluation of the appropriateness of the analytical models used, the applicability of experimental data used to validate the models, and the applicability of generic positions or those established in previous reviews of similar reactors.

   b. The reviewer determines the equipment, operating restrictions, and administrative procedures that are required to restrict possible control rod and bank worths, and the extent to which the alarm criterion in subsection I.3.b of this SRP section is satisfied. If the equipment involved is subject to frequent downtime, the reviewer must determine if alternative measures should be provided or the extent of proposed outage time is acceptable.

   c. The reviewer will employ the same procedures as in item 5.a, above, to evaluate the scram reactivity information described in subsection I.5 of this SRP section. The scram reactivity is a property of the reactor design and is not easily changed, but if restrictions are necessary the procedures in item 5.b, above, can be followed as applicable.

6. The information presented on criticality of fuel assemblies is reviewed in the context of the applicant''s physics calculations and the ability to calculate criticality of a small number of fuel assemblies. This information is related to information on fuel storage presented in SAR Section 9.1 and reviewed under SRP Section 9.1.1 and 9.1.2. The reviewer of SRP Section 9.1 assumes that teh applicant''s criticality calculations have been reviewed by CPB and are acceptable.

7. The reviewer exercises professional judgment and experience to ascertain the following about the applicant''s analytical methods:

   a. The computer codes used in the nuclear design are described in sufficient detail to enable the reviewer to establish that the theoretical bases, assumptions, and numerical approximations for a given code reflect the current state of the art.

   b. The source of the neutron cross-sections used in fast and thermal spectrum calculations is described in sufficient detail so that the reviewer can confirm that the cross-sections are comparable to those in the current ENDF/B data files (Ref. 4). If modifications and normalization of the cross-section data have been made, the bases used must be determined to be acceptable.

c. The procedures used to generate problem-dependent cross-section sets are given in sufficient detail so that the reviewer can establish that they reflect the state of the art. The reviewer confirms that the methods used for the following calculations are of acceptable accuracy: the fast neutron spectrum calculation; the computation of the U-238 resonance integral and correlation with experimental data; the computation of resonance integrals for other isotopes as appropriate (for example,Pu-240) calculation of the Dancoff correction factor for a given fuel lattice; the thermal neutron spectrum calculation; the lattice cell calculation including fuel rods, control assemblies, lumped burnable poison rods, fuel assemblies, and groups of fuel assemblies; and calculations of fuel and burnable poison depletion and buildup of fission products and transuranium isotopes.

d. The gross spatial flux calculations that are used in the nuclear design are discussed in sufficient detail so that the reviewer can confirm that the following items are adequate to produce results of acceptable accuracy; the method of calculation (e.g., diffusion theory, Sn transport theory, Monte Carlo, synthesis); the number of energy groups used; the number of spatial mesh intervals, when applicable; and the type of boundary conditions used, when applicable.

e. The calculation of power oscillations and stability indices for diametral xenon reactivity transients, axial xenon reactivity transients, other possible xenon reactivity transients, and non-xenon-induced reactivity transients, are discussed in sufficient detail so that the reviewer can confirm for each item that the method of calculation (e.g., nodal analysis, diffusion theory, transport theory, synthesis) and the number of spatial dimensions used (1, 2, or 3) are acceptable.

f. Verification of the data base, computer codes, and analysis procedures has been made by comparing calculated results with measurements obtained from critical experiments and operating reactors. The reviewer ascertains that the comparisons cover an adequate range for each item and that the conclusions of the applicant are acceptable.

8. The analysis of neutron irradiation of the reactor vessel may be used in two ways. It may provide the design basis for establishing the vessel material nil-ductility transition temperature as a function of the fluence, nvt. Or, it may provide the relative flux spectra at various positions between the pressure vessel and the reactor core so that the flux spectra for various test specimens may be estimated. This information is used in

determining the reactor vessel material surveillance program requirements and pressure-temperature limits for operation under SRP Sections 5.3.2 and 5.3.3. CPB reviews the calculational method, the geometric modeling, and the uncertainties in the calculational under this SRP section. The review procedures for pressure vessel irradiation include determinations that:

a. The calculations were performed by higher order theory than diffusion theory.

b. The geometric modeling is detailed enough to properly estimate the relative flux spectra at various positions from the reactor core boundary to the pressure vessel wall.

c. The peak vessel wall fluence for the design life of the plant is less than 1020 n/cm2 for neutrons of energy greater than 1 MeV. If the peak fluence is found to be greater than this value, the reviewers of SRP Sections 5.3.2 and 5.3.3 are notified.

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end.

topic 'II.D Evaluation Finding'.
say ('
4.3.IV Evaluation Finding

The reviewer verifies that sufficient information has been provided and his review supports the following type of evaluation finding, which is to be included in the staff''s safety evaluation report:

The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict experimental results. The staff concludes that the informa4ion presented adequately demonstrates the ability of these analysis to predict reactivity and physics characteristics of the _______________ plant.

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity requirements for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to shut down the reactor with at least a ______ %Wk/k subcritical margin in the hot condition at any time during the cycle with the highest worth control rod stuck in the fully withdrawn position.

On the basis of our review, we conclude that the applicant''s assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control requirements will be reviewed for additional cycles as this information becomes available.

The staff concludes that the nuclear design is acceptable and meets the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 11 with respect to prompt inherent nuclear feedback characteristics in the power operating range by:

   a. Calculating a negative Doppler coefficient of reactivity, and
   b. Using calculational methods that have been found acceptable.

   The staff has reviewed the Doppler reactivity coefficients in this case and found them to be suitably conservative.

2. The applicant has met the requirements of GDC 12 with respect to power oscillations which could result in conditions exceeding specified acceptable fuel design limits by:

   a. Showing that such power oscillations are not possible and/or can be easily detected and thereby remedied, and
   b. Using calculational methods that have been found acceptable.

   The staff has reviewed the analysis of these power oscillations in this case and found them to be suitably conservative.

3. The applicant has met the requirements of GDC 13 with respect to provision of instrumentation and controls to monitor variables and systems that can affect the fission process by:

   a. Providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables such as temperature and pressure, and
   b. Providing suitable alarms and/or control room indications for these monitored variables.

4. The applicant has met the requirements of GDC 26 with respect to provision of two independent reactivity control systems of different design by:

   a. Having a system than can reliably control anticipated operational occurrences,

   b. Having a system that can hold the core subcritical under cold conditions, and

c. Having a system that can control planned, normal power changes.

5. The applicant has met the requirements of GDC 27 with respect to reactivity control systems that have a combined capability in conjunction with poison addition by the emergency core cooling system of reliably controlling reactivity changes under postulated accident conditions by:

   a. Providing a movable control rod system and a liquid poison system, and

   b. Performing calculations to demonstrate that the core has sufficient shutdown margin with the highest worth stuck rod.

6. The applicant has met the requirements of GDC 28 with respect to postulated reactivity accidents by (reviewed by CPB under SRP Sections 15.4.8 or 15.4.9):

   a. Meetings the regulatory position in Regulatory Guide 1.77 for PWRs,

   b. Meeting the fuel enthalpy limit of 280 cal/gm for BWRs (same as for PWRs),

   c. Meeting the criteria on the capability to cool the core, and

   d. Using calculational methods that have been found acceptable for reactivity insertion accidents.

7. The applicant has met the requirements of GDC 10, 20, and 25 with respect to specified acceptable fuel design limits by providing analyses demonstrating:

   a. That normal operation, including the effects of anticipated operational occurrences, have met fuel design criteria,

   b. That the automatic initiation of the reactivity control system assures that fuel design criteria are not exceeded as a result of anticipated operational occurrences and assures the automatic operation of systems and components important to safety under accident conditions, and

   c. That no single malfunction of the reactivity control system causes violation of the fuel design limits.

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end.

topic 'II.E Implementation'.
say ('
4.3.V Implementation

The following is intended to provide guidance to applicants and licensees re-
garding the NRC staff''s plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alterna-
tive method for complying with specified portions of the Commission''s regula-
tions, the method described here in will be used by the staff in its evalua-
tion of conformance with Commission regulations.

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end.

topic 'II.F References'.
say ('
4.3.VI References

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design";
   Criterion 11, "Reactor Inherent Protection"; Criterion 12, "Suppression of
   Reactor Power Oscillations"; Criterion 13, "Instrumentation and Control";
   Criterion 20, "Protection System Functions"; Criterion 25, "Protection Sys-
   tem Requirements for Reactivity Control Malfunctions",; Criterion 26,
   "Reactivity Control System Redundancy and Capability"; Criterion 27, "Com-
   bined Reactivity Control Systems Capability"; and Criterion 28, "Reactivity
   Limits".

2. Branch Technical Position CPB 4.3-1, "Westinghouse Constant Axial Offset
   Control", July 1975, attached to SRP Section 4.3.

3. R. O. Meyer, "The Analysis of Fuel Densification", NUREG-0085, July 1976.

4. M. K. Drake, ed., "Data Formats and Procedures for the ENDF Neutron Cross
   Section Library", BNL-50274 (ENDF-102), National Neutron Cross Section
   Center, Brookhaven National Laboratory (1970).

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end.

topic 'II.G Appendix'.
say ('
4.3.VII Appendix

                    BRANCH TECHNICAL POSITION CPB 4.3-1
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# WESTINGHOUSE CONSTANT AXIAL OFFSET CONTROL (CAOC)

## A. BACKGROUND

In connection with the staff review of WCAP-8185 (17x17), we reviewed and accepted a scheme developed by Westinghouse for operating reactors that assures that throughout the core cycle including during the most limiting power maneuvers the total peaking factor, FQ, will not exceed the value consistent with the LOCA or other limiting accident analysis. This operating scheme, called constant axial offset control (CAOC), involves maintaining the axial flux difference within a narrow tolerance band around a burnup-dependent target in an attempt to minimize the variation of the axial distribution of xenon during plant maneuvers.

Originally (early 1974), the maximum allowable FQ (for LOCA) was 2.5 or greater. Later (late 1974), when needed changes were made to the ECCS evaluation model, Westinghouse, in order to meet physics analysis commitment to all its customers at virtually the same time, did a generic analysis (one designed to suit a spectrum of operating and soon-to-be-operating reac4ors) and showed that most plants could meet the requirements of Appendix K and 10 CFR 50.46 (i.e., 2200F peak clad temperature) if FQ F 2.32. Also, Westinghouse showed that CAOC procedures employing a +- 5% target band would limit peak FQ for each of these reactors to less than 2.32.

We recognized at that time, however, that not all plants needed to maintain FQ below 2.32 to meet FAC, or needed to operate within a +- 5% band to achieve FQ F 2.32. In fact, Point Beach was allowed to operate with a wider band because the Wisconsin Electric Power Company demonstrated to our satisfaction that the reactors could be maneuvered within a wider band (+6,-9%) and still hold FQ below 2.32. We fully expected that in time most plants would have individual CAOC analyses and procedures tailored to the requirements of their plant-specific ECCS analyses.

Therefore, when we accepted CAOC it was not just FQ = 2.32 and +-5% band width we were approving, but the CAOC methodology. This is analogous to our review and approval of ECCS and fuel performance evaluation models.

The CAOC methodology, which is described in Reference 1, entails (1) establishing an envelope of allowed power shapes and power densities, (2) devising an operating strategy for the cycle which maximizes plant flexibility (maneuvering) and minimizes axial power shape changes, (3) demonstrating that this strategy will not result in core conditions that violate the envelope of permissible core power characteristics, and (4) demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

Westinghouse argues that point 3, above, is achieved by calculating all of the load-follow maneuvers planned for the proposed cycle and showing that the maximum power densities expected are within limits. These calculations are performed with a radial/axial systhesis method which has been shown to predict conservative power densities when compared to experiment. While we have accepted CAOC on the basis of these analyses, we have also required that power distributions be measured throughout a number of representative (frequently, limiting) maneuvers early in cycle life to confirm that peaking factors are no greater than predicted.

Additionally, we are sponsoring a series of calculations at BNL to check aspects of the Westinghouse analysis.

The power distribution measurement tests described above will, of course, automatically relate incore and excore detector responses, and thereby validate that power distribution control can be managed with excore detectors.

B. BRANCH TECHNICAL POSITION

An applicant or licensee proposing CAOC for other than FQ = 2.32 and WI = +-5% is expected to provide:

1. Analyses of FQ x power fraction showing the maximum FQ (z) at power levels up to 100% and DNB performance with allowed axial shapes relative to the design bases for overpower and loss of flow transients. The envelope of these analyses must be shown to be valid for all normal operating modes and anticipated reactor conditions. (See Table 1 of References 2 for the cases which must be analyzed to form such an envelope.)

2. A description of the codes used, how cross-sections for cycle were determined, and what Fxy values were used.

3. A commitment to perform load-follow tests wherein FQ is determined by taking incore maps during the transient. (NOTE: Westinghouse has outlined for both the NRC staff and the ACRS an augmented startup test program designed to confirm experimentally the predicted power shapes. This program is presented in a Westinghouse report (Ref. 3). The tests will be carried out at several representative--both 15x15 and 17x17--reactors. We have endorsed these tests as has the ACRs in its June 12, 1975 letter for the Diablo Canyon plant. In addition, for the near term we plan to require that those licensees who propose to depart from the previously approved peaking factor and target band width perform similar tests, precisely which ones to be determined on a case-by-case basis, to broaden our confidence in analytical methods by extending the comparison of prediction with measurement to include more and more burnup histories.

C. REFERENCES

1. T. Morita, et al., "Power Distribution Control and Load Following Procedures", WCAP-8385 (proprietary) and WCAP-8403 (nonproprietary), Westinghouse

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   Electric Corporation, September 1974.

2. C. Eicheldinger, Westinghouse Electric Corporation, letter to D.B. Vassalo,
   U.S. Nuclear Regulatory Commission, July 16, 1975.

3. K. A. Jones, et al., "Augmented Startup and Cycle 1 Physics Program", WCAP-
   8575, Westinghouse Electric Corporation, August 1975.

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end.

topic 'III Thermal and Hydraulic Design'.
ask ('
4.4 THERMAL AND HYDRAULIC DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)
Secondary - Instrumentation and Control System Branch (ICSB)
            Human Factors Engineering Branch (HFEB)
            Procedures and Test Review Branch (PTRB)

(please click on your choice)

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[
'4.4.I Areas of Review',
'4.4.II Acceptance Criteria',
'4.4.III Review Procedures',
'4.4.IV Evaluation Findings',
'4.4.V Implemetation',
'4.4.VI References',
'Content',
'Index',
'Opening Menu']).

do (?part).
end.

topic 'III.A Areas of Review'.
say ('
4.4.I Areas of Review

The objectives of the review are to confirm that the thermal and hydraulic de-
sign of the core and the reactor coolant system (RCS) has been accomplished
using acceptable analytical methods; is equivalent to or is a justified extra-
polation from proven designs; provides acceptable margins of safety from con-
ditions which would lead to fuel damage during normal reactor operation and
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anticipated operational transients; and is not susceptible to thermal-hydraulic design, i.e., that for a plant similar in core and primary coolant system design to previously reviewed plants. The review of new prototype plants, new CHF or CPR correlations, and new analysis methods require that additional independent audit analyses be performed. The required analyses may be in the following form:

1. Independent computer calculations to substantiate reactor vendor analyses.

2. Reduction and correlations of experimental data to verify processes or phenomena which are applied to reactor design.

3. Independent comparisons and correlations are made of data from experimental programs. These reviews also include analyses of experimental techniques, test repeatability, and data reduction methods.

The review includes evaluation of the proposed technical specifications regarding safety limits and limiting safety settings, to ascertain that these are consistent with the power-flow operating map for boiling water reactor (BWR) plants or the temperature-power operating map for pressurized water reactor (PWR) plants.

The review also includes determination of the largest hydraulic loads on core and reactor coolant system components during normal operation and postulated accident conditions. This information is used in the review of fuel holddown requirements.

To accomplish the objectives, the reviewer examines features of core and RCS components, key process variables for the coolant system, calculated parameters characterizing thermal performance, data serving to support new correlations or changes in accepted correlations, and assumptions in the equations and solution techniques used in the analyses. The reviewer determines that the applicant has used approved analysis methods in the manner specified by topical reports describing the methods and by staff reports approving the methods. the analysis methods addressed are to include core thermal-hydraulic calculations to establish local coolant conditions, departure from nucleate boiling or boiling transition calculations, and thermal-hydraulic stability evaluation. If an applicant has used previously unapproved correlations or analysis methods, the reviewer initiates an evaluation, either generic or plant specific. Any changes to accepted codes, correlations, and analytical procedures, or the addition of new ones must be reviewed to determine that they are justified on theoretical or empirical grounds.

A secondary review is performed by ICSB, HFEB, and PTRB. ICSB will review the functional performance and requirements for Inadequate Core Cooling (ICC) monitoring system hardware. Emergency procedure guidelines and the information display will be reviewed by PTRB and HFEB, respectively. The results of these reviews will be used by CPB to complete the overall evaluation of the thermal-hydraulic review and will be incorporated into the Safety Evaluation Report (SER).

The review of power distribution assumption made for the core thermal and hydraulic analysis is coordinated with the review for core physics calculations, as described in the Standard Review Plan (SRP) Section 4.3, for consistency. The reviewer verifies that core monitoring techniques which rely on in-core or ex-core neutron sensor inputs are reviewed.

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topic 'III.B Acceptance Criteria'.
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4.4.II Acceptance Criteria

The CPB acceptance criteria are based meeting the relevant requirements of General Design Criterion 10 (Ref. 1), as it relates to the reactor core being designed, with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operation or anticipated operational occurrences (AOO).

Specific criteria necessary to meet the requirements of GDC 10 are as follow:

1. SRP Section 4.2 specifies the acceptance criteria for evaluation of fuel design limits. One of the criteria provides assurances that there be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) or transition condition during normal operation or anticipated operational occurrence.

   Uncertainties in the values of process parameters, core design parameters, and calculational methods used in the assessment of thermal margin should be treated with at least a 95% probability at a 95% confidence level.

   Two examples of acceptable approaches to meet this criterion are:

   a. For departure from nucleate boiling ration (DNBR), critical heat flux ratio (CHFR) or critical power ratio (CPR) correlations there should be at 95% probability at the 95% confidence level, that the hot rod in the core does not experience a departure from nucleate boiling or boiling transition condition during normal operation or anticipated operational occurrences; or

   b. For DNBR, CHFR or CPR correlations, the limiting (minimum) value of DNBR, CHFR, of CPR is to be established such that at least 99.9% of the fuel rods in the core would not be expected to experience departure from nucleate boiling or boiling transition during normal operation or anticipated operational occurrences.

Correlations of critical heat flux are continually being revised as a result of additional experimental data, changes in fuel assembly design, and improved calculational techniques involving coolant mixing and the effect of axial power distributions. As guidance to the reviewer, the correlations listed below have been found acceptable for previously reviewed plants.

a. BWRs - The value of the minimum CPR calculated with the GETAB analysis (Ref. 2) will vary for different plants and/or fuel types. Typical values are 1.06 and 1.07.

b. PWRs - The value of the minimum DNBR calculated with due allowance for mixing grids (Refs. 3,4, and 5) is typically 1.30 using the BAW-2 correlation (Ref. 6) or the W-3 correlation (Ref. 7). Much lower values, depending upon the test data base and fuel design, are acceptable for more recent correlations such as the WRB-1, CE-1, and BWC.

2. Problems affecting DNBR or CPR limits, such as fuel densification or rod bowing, are accounted for by an appropriate design penalty which is determined experimentally or analytically. Subchannel hydraulic analysis codes such as those described in References 8 and 9, should be used to calculate local fluid conditions within fuel assemblies for use in PWR DNB correlations. The acceptability of such codes must be demonstrated by measurements made in large lattice experiments or power reactor cores. The effects of radial pressure gradients in the core flow distribution should be evaluated. Calculations of BWR fluid conditions for use in CHF correlations have been in accordance with the models specified in Reference 10 and 11.

3. The reactor should be demonstrated to have sufficient margin to be free of undamped oscillations and other thermal-hydraulic instabilities for all conditions of steady-state operation (including part loop operation), and for anticipated operational occurrences.

4. Methods for calculating single-phase and two-phase fluid flow in the reactor vessel and other components should include classical fluid mechanics relationships and appropriate empirical correlations. For components of unusual geometry, such as the following, these relationships should be confirmed empirically, using representative data bases from approved reports of the type listed below.

   a. Reactor vessel (Ref. 12).

   b. Jet pump (Ref. 13).

   c. Core flow distribution (Refs. 12 and 14).

5. The proposed technical specifications should be established such that the

plant can be safely operated at steady state conditions under all of the expected combinations of system parameters. The safety limits and limiting safety settings must be established for each parameter, or combinations of parameters, such that acceptance criterion 1, above, is satisfied.

6. Preoperational and initial startup test programs should follow the recommendations of Regulatory Guide 1.68 (Ref. 15), as regards measurements, and confirmation of thermal hydraulic design aspects.

7. The design description and proposed procedures for use of the loose part monitoring system should be consistent with the requirements of Regulatory Guide 1.133 (Ref. 16).

8. The effects of crud should be accounted for in the thermal-hydraulic design by including it in the CHF calculations in the core or in the pressure drop throughout the RCS. Process monitoring provisions should assure capability for detection of a three percent pressure drop in the reactor coolant flow. The flow should be monitored every 24 hours.

9. Instrumentation provided for an unambiguous indication of ICC, such as primary coolant saturation meters in PWRs, reactor vessel measurement systems, and core exit thermocouples, should meet the design requirements described in item II.F.2 NUREG-0718 (Ref. 17) and NUREG-0737 (Ref. 18). Procedures for detection and recovery from conditions of ICC must be consistent with technical guidelines that incorporate response predictions based on appropriate analyses.

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end.

topic 'III.C Review Procedures'.
say ('
4.4.III Review Procedures

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in section II of this SRP section. For operating license (OL) applications, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to safety limits, limiting safety system settings, and conditions of operation.

The reviewer must begin with an understanding of currently acceptable thermal and hydraulic design practice for the reactor type under review. This understanding can be most readily gained from topical reports describing CHF correlations, system hydraulic models and tests, and core subchannel analysis methods; from standard texts and other technical literature which establish the methodology and the nomenclature of this technology; and from documents which summarize current staff positions concerning acceptable design methods.

Much of the review described below is generic in nature and is not performed for each plant. That is, the CPB reviewer is to compare the core design and operating parameters to those of previously reviewed plants. He then devotes the major portion of his review effort to those areas where the application is not identical to previously reviewed plants.

The reviewer is to compare the information in the applicant''s safety analysis report (SAR) to the documents referenced by the applicant or in this SRP Section to determine conformance to the bounds established by such documents. The reviewer must confirm that void, pressure drop, and heat transfer correlations used to estimate fluid conditions (flow, pressure, quality) are within the ranges of applicability specified by their authors or in previous staff reviews, that the analysis methods are used in the manner specified by the developers or in previous staff reviews, that the reactor design falls within the ranges of applicability specified for accepted analysis methods, and that the design is within the criteria specified in II, above, and is not an unexplained or unwarranted extrapolation of other thermal-hydraulic designs.

The review does not routinely involve calculations by the staff. However, the reviewer should ensure that those applications based on statistical design methodologies include the coefficients require by the statistical model and define the parameter ranges for which the coefficients are applicable. Uncertainties in computer codes, correlations, design methods, and set point methodologies should be quantified and the method(s) of accounting for these uncertainties in the design procedures should be discussed. For example the sensitivity factors and their ranges of applicability must be reviewed for those plants utilizing the Westinghouse "Improved Thermal Design Procedure", (Ref. 19). On occasion, e.g., if a new design or new design method is proposed, independent analyses are performed by the staff or by consultant under the direction of CPB. These analyses verify the design or establish the range of applicability and associated accuracy of the new method and the reviewer ensures it is applied accordingly.

The reviewer is to establish that the thermal-hydraulic design and its characterization by MCHFR or DNBR have been accomplished and are presented in a manner which accounts for all possible reactor operating states as determined from operating maps. In this regard, the reviewer must confirm that the power distribution assumptions of SAR Section 4.4 are a conservative (i.e., worst-case) accounting of the power distributions derived in SAR Section 4.3 from core physics analyses, and that the latter analyses include an acceptable calculation of local void fractions. He must also confirm that the mass flux used in these calculations takes into account the core flow distribution (including that for partial loop operation) and the worst case of core bypass flow. The reviewer confirms that the primary coolant flow range shown in the operating map will be verified by prestartup measurements.

The reviewer ensures that adequate account is taken of the effect of crud in the primary coolant system, such as in the calculation of CHF in the core, heat transfer in the steam generators, and pressure drop throughout the RCS.

The reviewer is to examine the calculation of hydraulic loads for normal operations, including anticipated transients, to ensure they are properly estimated for the worst cases. Worst case hydraulic loads for normal operations are to be provided for use in the analysis of lifting force of the fuel (SRP Section 4.2). CPB will also provide consultation to RSB upon request, regarding calculations for postulated accident conditions. MEB reviews the adequacy of components and structures under accident loads (SRP Section 3.9.2) and CPB determines that a coolable core geometry is maintained (SRP Section 4.2).

The reviewer should ensure that an adequate loose parts monitoring system is provided. At the CP level, the design criteria for the system and the types, locations, and methods of mounting all intended sensors should be reviewed. The reviewer should compare the design to Regulatory Guide 1.33 and to equipment used and application experience on comparable plants.

At the OL level; a more complete description of the system including sensitivity specifications and operating procedures should be reviewed. The reviewer should ensure that operating procedures and training provisions are adequate to fully utilize the system potential for loose parts detection. The Operator Licensing Branch (OLB) will provide consultation on staff training in accordance with the SRP Section 13.2.

The reviewer should review the vibration monitoring equipment and procedures to ensure that the monitoring provisions are adequate for the plant under review based on experience with comparable plants. The CPB will evaluate the application of neutron monitoring sensors for core vibration test analysis. The MEB is responsible in SRP Section 3.9.2, and provides technical consultation to CPB on the need for permanent vibration monitoring provisions for the plant under review.

The reviewer ensures that applicants have an acceptable program for incorporation of instrumentation and procedures for detection and recovery from conditions of inadequate core cooling. At the CP stage, the applicant must provide preliminary design information on selected instrumentation components and must specify the design concept selected for development instrumentation in accordance with the requirement of item II.F.2 of NUREG-0718 (Ref. 17).

At the OL stage, the reviewer ensures that the applicant is in compliance with the documentation requirements and design requirements described in item II.F.2 of NUREG-0737 (Ref. 18). The reviewer consults with ICSB and HFEB concerning the design acceptability of the instrumentation and displays with the Reactor Systems Branch (RSB) and PTRB concerning the acceptability of guidelines and procedures for recognition and response to inadequate core cooling conditions.

The applicant''s proposed preoperational and initial startup test programs are reviewed to determine that they are consistent with the intent of Regulatory Guide 1.68 (Ref. 15). At the OL stage, the reviewer is to assure that sufficient information is provided by the applicant to identify clearly the test objectives, methods of testing, and acceptance criteria. (See part C.2.b of Reference 15).

The test scope should include verification of any safety analysis codes or methods which could affect the thermal-hydraulic evaluations and which have not been previously verified. The initial startup test should also include a description of plans for a signature analysis to determine alarm setting for the loose parts monitoring system, and a description of test programs for evaluation, qualification and calibration of ICC instrumentation.

The reviewer evaluates the proposed test programs to determine if they provide reasonable assurance that the core and reactor coolant system will satisfy functional requirements. As an alternative to this detailed evaluation, the reviewer may compare the core and reactor coolant system design to that of previously reviewed plants. If the design is essentially identical and if the proposed test programs are essentially the same as performed previously on other plants, the reviewer may conclude that the proposed test programs are adequate for the core and reactor coolant system.

If the core the reactor coolant system differs significantly from that of previously reviewed designs, the impact of the proposed changes on the preoperational and initial startup testing programs are reviewed at the construction permit stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

The proposed technical specifications that relate to the core and the reactor coolant system are evaluated. This evaluation is to cover all of the safety limits and bases that could affect the thermal and hydraulic performance of the core. The limiting safety system settings are reviewed to ascertain that acceptable margins exist between the values att which reactor trip occurs automatically for each parameter (or combinations of parameters) and the safety limits. The reviewer confirms that the limiting safety system settings and limiting conditions for operation, as they relate to the reactor coolant system, do not permit operation with any expected combination of parameters that would not satisfy criterion 1 of section II. For example, the limiting condition of operation must assure that the reactor coolant pumps have adequate net positive suction head all expected modes of operation.

(please press Space to continue)').
end.

topic 'III.D Evaluation Findings'.
say ('
4.4.IV Evaluation Findings

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff''s safety evaluation report. The following paragraph is applicable to both a CP and OL:

> The thermal-hydraulic design of the core for the _________ plant was reviewed. The scope of review included the design criteria, preliminary core design, and the steady state analysis of the core thermal-hydraulic performance. (For an OL review this sentence should be modified to include the implementation of the design criteria as represented by the final core design.) The review concentrated on the differences between the proposed core design (and criteria) and those designs and criteria that have been previously reviewed and found acceptable by the staff. It was found that all such differences were satisfactorily justified by the applicant. The applicant''s thermal-hydraulic analyses were performed using analytical methods and correlations that have been previously reviewed by the staff and found acceptable.

For a CP, the following conclusions should be made:

> The staff concludes that the thermal-hydraulic design of the core meets the requirements of General Design Criterion 10, 10 CFR Part 50. The core has been designed with appropriate margin to assure that acceptable fuel design limits are not exceeded during steady-state operation or anticipated operational occurrences. This conclusion is based on the applicant''s analyses of the core thermal-hydraulic performance which was reviewed by the staff and found to be acceptable. The applicant will establish a preoperational and initial startup test program in accordance with Regulatory Guide 1.68 to measure and confirm the thermal-hydraulic design aspects. The loose parts and vibration monitoring system is designed for compliance with the requirements of Regulatory Guide 1.133 and the instrumentation for the detection of inadequate core cooling is in compliance with the requirements of item II.F.2 of NUREG-0718.

For an OL application, the following types of conclusions should be supported.

> The staff concludes that the thermal-hydraulic design of the core meets the requirements of General Design Criterion 10, 10 CFR Part 50 and is acceptable for final design approval. We also conclude that the reactor core has been designed with appropriate margin to assure that the reactor fuel design limits are not exceeded during steady-state operation or anticipated operational occurrences and that the reactor will perform its safety functions throughout its design lifetime under all modes of operation. This conclusion is based on the applicant''s analyses of the core

thermal-hydraulic performance which was reviewed by the staff and found to be acceptable. The applicant has committed to a preoperational and initial startup test program in accordance with Regulatory Guide 1.68 to measure and confirm the thermal-hydraulic design aspects. The staff has reviewed the applicant''s preoperational and initial startup test program and has concluded that it is acceptable. We also conclude that the loose parts monitoring program is designed for compliance with the requirements of Regulatory Guide 1.133, and is, therefore, acceptable. We have reviewed the instrumentation for the detection of inadequate core cooling and concluded that it is in compliance with the acquirements of Item II.F.2 of NUREG-0737 and is acceptable.

(please press Space to continue)').
end.

topic 'III.E Implementation'.
say ('
4.4.V Implementation

The following is intended to provide guidance to applicants and licensees regarding the NRC staff''s plan for using this SRP section.

Except in those cases which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission''s regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGs.

(please press Space to continue)').
end.

topic 'III.F References'.
say ('
4.4.VI References

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design".

2. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application", NEDO-10958, General Electric Company (1973).

3. F. F. Cadek, F. E. Motley, and D. P. Dominicis, "Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid", WCAP-7941-L (proprietary), Westinghouse Electric Corporation, June 1972.

4. F. E. Motley and F. F. Cadek, "DNB Test Results for New Mixing Vane Grids

(R), "WCAP-7695-L (proprietary), Westinghouse Electric Corporation, July 1972.

5. F. E. Motley and F. F. Cadek, "Application of Modified Spacer Factor to L Grid Typical and Cold Wall Cell DNB", WCAP-7988, Westinghouse Electric Corporation, October 1972. (See also WCAP-8030.)

6. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water", in "Two-Phase Flow and Heat Transfer in Rod Bundles", American Society of Mechanical Engineers, New York (1969). (See also BAW-10000 and BAW-10036.)

7. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution", Journal of Nuclear Energy, Vol. 21, 241-248 (1967).

8. "TEMP - Thermal Enthalpy Mixing Program", BAW-10021, Babcock and Wilcox Company, April 1970.

9. H. Chelemer, P. T. Chu, and L. E. Hochreiter, "THINC-IV-An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores", WCAP-7956, Westinghouse Electric Corporation, June 1973. (See also WCAP-7359-L and WCAP-7838)

10. B. C. Slifer and J. E. Hench, "Loss of Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors", NEDO-10329, Appendix C, General Electric Company, April 1971.

11. J. Duncan and P. W. Marriott, "General Electric Company Analytical Model for Loss of Coolant Accident Analysis in Accordance with 10 CFR Part 50, Appendix K", NEDO-20566, General Electric Company, November 1975.

12. B. S. Mullanax, R. J. Walker and B. A. Karrasch, "Reactor Vessel Model Flow Tests", BAW-10037 (non-proprietary version of BAW-10012), Revision 2, Babcock and Wilcox Company, September 1968.

13. "Design and Performance of General Electric Boiling Water Reactor Jet Pumps", APED-5460, General Electric Company, September 1968.

14. H. T. Kim, "Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello", NEDO-10299, General Electric Company, January 1971.

15. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors".

16. Regulatory Guide 1.133, "Loose Parts Detection Program for the Primary System of Light-Water-Cooled Reactors".

17. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits".

18. NUREG-0737, "Clarification of TMI Action Plant Requirements".

19. H. Chelemer, L. H. Boman, D. R. Sharp, "Improved Thermal Design Procedure" WCAP-8567(P)/8568(NP), Westinghouse Electric Corporation, July 1975.

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(please press Space to continue)').
end.

topic 'Criterion 10'.
window ().
say ('
Criterion 10 - Reactor Design

The reactor core and associated coolant, control, and protection
systems shall be designed with appropriate margin to assure that
specified acceptable fuel design limits are not exceeded during
any condition of normal operation, including the effects of
anticipated operational occurrences.').
close_window ().
end.

topic 'GDC 27 and 35'.
do ('Criterion 27') and do ('Criterion 35').
end.

topic 'Criterion 27'.
window ().
say ('
Criterion 27 - Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a
combined capability, in conjunction with poison addition by the
emergency core cooling system, of reliably controlling reacti-
vity changes to assure that under postulated accidents conditi-
ons and with appropriate margin for stuck rods the capability to
cool the core is maintained.').
close_window ().
end.

topic 'Criterion 35'.
window ().
say ('
Criterion 35 - Emergency core cooling
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A system to provide abundant emergency core cooling to provide
abundant emergency core cooling shall be provided. The system
safety function shall be to transfer heat from the reactor core
following any loss of reactor coolant at a rate such that (1)
fuel and clad damage that could interfere with continued effec-
tive core cooling is prevented and (2) clad metal-water reaction
is limited to negligible amounts.

Suitable redundancy in components and features, and suitable
interconnections, leak detection, isolation, and containment
capabilities shall be provided to assure that for onsite electric
power system operation (assuming onsite power is not available)
the system safety function can be accomplished, assuming a
single failure.').
close_window ().
end.

topic 'General Design Criteria 27 and 35'.
do ('Criterion 27') and do ('Criterion 35').
end.

topic 'General Design Criteria 10, 27, and 35'.
do ('Criterion 10') and do ('Criterion 27') and do ('Criterion 35').
end.

topic 'General Design 2'.
window ().
say ('
Criterion 2 - Design Bases for Protection against Natural
Phenomena

Structures, systems, and components important to safety shall be
designed to withstand the effect of natural phenomena such as
earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches
without loss of capability to perform their safety functions. The
design bases for these structures, systems, and components shall
reflect: (1) Approriate consideration of the most severe of the
natural phenomena that have been historically reported for the
site and surrounding area, with sufficient margin for the limited
accuracy, quantity, and period of time in which the historical
data have been accumulated, (2) appropriate combinations of the
effects of normal and accident conditions with the effects of the
natural phenomena and (3) the importance of the safety functions
to be performed.').
close_window ().
end.
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topic 'Criterion 11'.
window ().
say ('
Criterion 11 - Reactor Inherent Protection

The reactor core and associated coolant systems shall be designed
so that in the power operating range the net effect of the prompt
inherent nuclear feedback characteristics tends to compensate for
a rapid increase in reactivity.').
close_window ().
end.

topic 'Criterion 12'.
window ().
say ('
Criterion 12 - Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection
systems shall be designed to assure that power oscillations which
can result in conditions exceeding specified acceptable fuel de-
sign limits are not possible or can be reliably and readily detec-
ted and suppressed.').
close_window ().
end.

topic 'Criterion 13'.
window ().
say ('
Criterion 13 - Instrumentation and control

Instrumentation shall be provided to monitor variables and systems
over their anticipated ranges for normal operation, for anticipated
operational occurrences, and for accident conditions as appropriate
to assure adequate safety, including those variables and systems
that can affect the fission proccess, the integrity of the reactor
core, the reactor coolant pressure boundary, and the containment
and its associated systems. Appropriate controls shall be provided
to maintain these variables and systems within prescribed operating
ranges.').
close_window ().
end.

topic 'Criterion 20'.
window ().
say ('
Criterion 20 - Protection System Functions

The protection system shall be designed (1) to initiate automati-
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cally the operation of appropriate systems including the reactivity
control systems, to assure that specified acceptable fuel design
limits are not exceeded as a result of anticipated operational
occurrences and (2) to sense accident conditions and to initiate
the operation of systems and components important to safety.').
close_window ().
end.

topic 'Criterion 25'.
window ().
say ('
Criterion 25 - Protection System Requirements for Reactivity
Control Malfunctions.

The protection system shall be designed to assure that specified
acceptable fuel design limits are not exceeded for any single mal-
function of the reactivity control systems, such as accidental
withdrawn (not ejection or dropout) of control ro$s.').
close_window ().
end.

topic 'Criterion 26'.
window ().
say ('
Criterion 26 - Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design
principles shall be provided. One of the systems shall use control
rods, preferably including a positive means for inserting the rods,
and shall be capable of reliably controlling reactivity changes to
assure that under conditions of normal operation, including anti-
cipated operational occurrences, and with appropriate margin for
malfunctions such as stuck rods, specified acceptable fuel design
limits are not exceeded. The second reactivity control system shall
be capable of reliably controlling the rate of reactivity changes
(including xenon burnout) to assure acceptable fuel design limits
are not exceeded. One of the systems shall be capable of holding the
reactor core subcritical under cold conditions.').
close_window ().
end.

topic 'Criterion 28'.
window ().
say ('
Criterion 28 - Reactivity Limits

The reactivity control systems shall be designed with appropriate
limits on the potential amount and rate of reactivity increase to
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assure that the effects of postulated reactivity accidents can
neither (1) result in damage to the reactor coolant pressure boun-
dary greater than limited local yielding nor (2) sufficiently dis-
turb the core, its support structures or other reactor pressure
vessel internals to impair significantly the capability to cool the
core. These postulated reactivity accidents shall include conside-
ration of rod ejection (unless prevented by positive means), rod
dropout, steam line rupture, changes in reactor coolant temperature
and pressure, and cold water addition.').
close_window ().
end.

topic 'General Design Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28'.
do ('Criterion 10') and do ('Criterion 11') and do ('Criterion 12') and do
('Criterion 13') and do ('Criterion 20') and do ('Criterion 25') and do
('Criterion 26') and do ('Criterion 27') and do ('Criterion 28').
end.

topic 'SRP Section 4.3'.
do ('4.3 Nuclear Design').
end.

topic 'SRP Section 4.4'.
do ('4.4 Thermal and Hydraulic Design').
end.


end.
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